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

May 16, 2013

Mr. Steven D. Capps
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME8213 AND ME8214)

Dear Mr. Capps:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 269 to Renewed Facility Operating License NPF-9 and Amendment No. 249 to Renewed Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2 (McGuire 1 and 2). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 5, 2012, as supplemented by letters dated May 29, 2012, June 21, 2012, July 6, 2012, July 16, 2012, August 15, 2012, September 27, 2012, November 1, 2012, January 2, 2013, and March 7, 2013. The amendments revise the TSs to implement a measurement uncertainty recapture power uprate at McGuire 1 and 2.

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

If you have any questions, please call me at 301-415-2901.

Sincerely,

John P. Boska, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 269 to NPF-9
2. Amendment No. 249 to NPF-17
3. Safety Evaluation

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 269
Renewed License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9, filed by the Duke Energy Carolinas, LLC (the licensee), dated March 5, 2012, as supplemented by letters dated May 29, 2012, June 21, 2012, July 6, 2012, July 16, 2012, August 15, 2012, September 27, 2012, November 1, 2012, January 2, 2013, and March 7, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 269 , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the completion of the facility's end-of-cycle 23 refueling outage currently scheduled for the fall of 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-9
and the Technical Specifications

Date of Issuance: May 16, 2013

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. NPF-17, filed by the Duke Energy Carolinas, LLC (the licensee), dated March 5, 2012, as supplemented by letters dated May 29, 2012, June 21, 2012, July 6, 2012, July 16, 2012, August 15, 2012, September 27, 2012, November 1, 2012, January 2, 2013, and March 7, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 249 , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the completion of the facility's end-of-cycle 22 refueling outage currently scheduled for the spring of 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-17
and the Technical Specifications

Date of Issuance: May 16, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 269

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 249

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

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 Pages	License Pages
NPF-9, page 3	NPF-9, page 3
NPF-9, page 4	NPF-9, page 4
NPF-17, page 3	NPF-17, page 3
NPF-17, page 4	NPF-17, page 4
TS Pages	TS Pages
1.1-5	1.1-5
3.7.1-3	3.7.1-3
Appendix B Pages	Appendix B Pages
-	NPF-9, page B-3
-	NPF-17, page B-3

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2, and;

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

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

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

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3469 megawatts thermal (100%).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No.269, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Updated Final Safety Analysis Report</u>

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than June 12, 2021, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Fire Protection Program

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SER dated March 1978 and Supplements 2, 5 and 6 dated March 1979, April 1981, and February 1983, respectively, and the safety evaluation dated May 15, 1989, subject to the following provision:

> Duke may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No.269 , are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

(6) Antitrust Conditions

The licensee shall comply with the antitrust conditions delineated in Appendix C of this renewed operating license.

(7) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

A) Fire fighting response strategy with the following elements:
   1. Pre-defined coordinated fire response strategy and guidance
   2. Assessment of mutual aid fire fighting assets
   3. Designated staging areas for equipment and materials
   4. Command and control
   5. Training of response personnel

B) Operations to mitigate fuel damage considering the following:
   1. Protection and use of personnel assets
   2. Communications
   3. Minimizing fire spread
   4. Procedures for implementing integrated fire response strategy
   5. Identification of readily-available pre-staged equipment
   6. Training on integrated fire response strategy
   7. Spent fuel pool mitigation measures

C) Actions to minimize release to include consideration of:
   1. Water spray scrubbing
   2. Dose to onsite responders

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3469 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.249 , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59, and otherwise complies with the requirements in that section.

(4) Fire Protection Program

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SER dated March 1978 and Supplements 2, 5, and 6 dated March 1979, April 1981, and February 1983, respectively, and the safety evaluation dated May 15, 1989, subject to the following provisions:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Protection of the Environment

Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement dated April 1976, the licensee shall provide written notification to the Office of Nuclear Reactor Regulation.

(6) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No.249 , are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

(7) Antitrust Conditions

The licensee shall comply with the antitrust conditions delineated in Appendix C of this renewed operating license.

(8) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

A) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

B) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.*
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.

* Following implementation of MUR on the respective Unit, the value of RTP shall be 3469 MWt.

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power Range Neutron Flux High
Setpoints in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINTS (% RTP)
4	≤ 57
3	≤ 38
2	≤ 19

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± 3%)
STEAM GENERATOR				
A	B	C	D	
SV-20	SV-14	SV-8	SV-2	1170
SV-21	SV-15	SV-9	SV-3	1190
SV-22	SV-16	SV-10	SV-4	1205
SV-23	SV-17	SV-11	SV-5	1220
SV-24	SV-18	SV-12	SV-6	1225

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-9

Duke Energy Carolinas, LLC comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
	The Licensee shall perform an analysis, in the form of either a topical report or site-specific analysis, describing how the current P-T limit curves at 34 Effective Full Power Years (EFPY) for McGuire Unit 1 and the methodology used to develop these curves considered all Reactor Vessel (RV) materials (beltline and non-beltline) and the lowest service temperature of all ferritic Reactor Coolant Pressure Boundary (RCPB) materials, as applicable, consistent with the requirements of 10 CFR Part 50, Appendix G. This analysis shall be provided to the NRC within one year after NRC approval of the March 5, 2012 McGuire Measurement Uncertainty Recapture (MUR) License Amendment Request.	See Condition
	McGuire Nuclear Station switchyard voltages required (so as not to impact the degraded voltage relay settings), corresponding to Unit 1 post-MUR uprate conditions, will be evaluated prior to implementation of MUR on Unit 1. However, if at the time of this evaluation, Unit 1 is not capable of realizing the expected maximum post-MUR uprate MWt power level and/or Unit 1 is not capable of generating the expected maximum post-MUR uprate MWe, then an additional evaluation will be performed when Unit 1 has these capabilities. If this additional evaluation is necessary, any changes in the switchyard voltages required (so as not to impact the degraded voltage relay settings), corresponding to conditions associated with the additional Unit 1 MWt capability and/or the additional Unit 1 MWe capability, will be evaluated prior to raising Unit 1 reactor core full steady state power to the expected maximum post-MUR uprate MWt power level and/or prior to Unit 1 generating the expected maximum post-MUR uprate MWe.	See Condition

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-17

Duke Energy Carolinas, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
	The Licensee shall perform an analysis, in the form of either a topical report or site-specific analysis, describing how the current P-T limit curves at 34 Effective Full Power Years (EFPY) for McGuire Unit 2 and the methodology used to develop these curves considered all Reactor Vessel (RV) materials (beltline and non-beltline) and the lowest service temperature of all ferritic Reactor Coolant Pressure Boundary (RCPB) materials, as applicable, consistent with the requirements of 10 CFR Part 50, Appendix G. This analysis shall be provided to the NRC within one year after NRC approval of the March 5, 2012 McGuire Measurement Uncertainty Recapture (MUR) License Amendment Request.	See Condition

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 269 TO RENEWED FACILITY OPERATING LICENSE NPF-9

AND

AMENDMENT NO. 249 TO RENEWED FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By application dated March 5, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12082A210), as supplemented by letters dated May 29, 2012 (ADAMS Accession No. ML12160A085), June 21, 2012 (ADAMS Accession No. ML12187A174), July 6, 2012 (ADAMS Accession No. ML12199A023), July 16, 2012 (ADAMS Accession No. ML12209A175), August 15, 2012 (ADAMS Accession No. ML12250A622), September 27, 2012 (ADAMS Accession No. ML12284A130), November 1, 2012 (ADAMS Accession No. ML12310A384), January 2, 2013 (ADAMS Accession No. ML13024A406), and March 7, 2013 (ADAMS Accession No. ML13079A330), Duke Energy Carolinas, LLC (Duke Energy, the licensee), requested changes to the Technical Specifications (TSs) for the McGuire Nuclear Station, Units 1 and 2 (McGuire 1 and 2). The supplements dated May 29, 2012, June 21, 2012, July 6, 2012, July 16, 2012, August 15, 2012, September 27, 2012, November 1, 2012, January 2, 2013, and March 7, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 15, 2012 (77 FR 28630).

The proposed changes would revise the TSs to implement a measurement uncertainty recapture (MUR) power uprate at McGuire 1 and 2. This amendment would raise the reactor thermal power (RTP) from 3411 megawatts-thermal (MWt) to 3469 MWt upon implementation.

Enclosure

2.0 BACKGROUND

2.1 Measurement Power Uncertainty Recapture Power Uprates

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

A change to the Commission's regulations at 10 CFR Part 50, Appendix K, was published in the *Federal Register* on June 1, 2000 (65 FR 34913), which became effective July 31, 2000. This change allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not a power level less than the licensed power level, based on the use of state-of-the art feedwater (FW) flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided that the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. As there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, this change to 10 CFR 50, Appendix K, did not authorize increases in licensed power levels for individual nuclear power plants. As the licensed power level for a plant is contained in its operating license, licensees seeking to raise the licensed power level must submit a license amendment request (LAR) which must be reviewed and approved by the NRC staff. McGuire 1 and 2 is currently licensed to operate at a maximum power level of 3411 MWt, which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in RTP measurement. The LAR would reduce this uncertainty to 0.3 percent.

In order to provide guidance to licensees seeking a MUR power uprate on the basis of improved FW flow measurement, the NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (ADAMS Accession No. ML013530183). RIS 2002-03 provides guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR power uprate LARs. While RIS 2002-03 does not constitute an NRC requirement, its use aids licensees in the preparation of their MUR power uprate LAR, while also providing guidance to the NRC staff for the conduct of its review. The licensee stated in its application dated March 5, 2012, that its LAR was submitted consistent with the guidance of RIS 2002-03.

2.2 Implementation of an MUR Power Uprate at McGuire 1 and 2

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

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. However, error 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.

The licensee intends to use UFMs developed by the Cameron International Corporation (Cameron, formerly known as Caldon Ultrasonic Inc. (Caldon)), the leading edge flow meter (LEFM) CheckPlus System, which provides a more accurate measurement of FW flow as compared to the accuracy of the venturi flow meter-based instrumentation originally installed at McGuire 1 and 2. Installation of these UFMs to measure FW flow would allow the licensee to operate the plant with a reduced instrument uncertainty margin and an increased power level in comparison to its currently licensed thermal power (CLTP).

The Cameron LEFM CheckPlus System was developed over a number of years. Cameron submitted a topical report in March of 1997, ER-80P, 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. This topical report also provides guidance for determining plant-specific power calorimetric uncertainties. The NRC staff approved the use of this topical report in a safety evaluation (SE) dated March 8, 1999 (ADAMS Accession No. ML11353A017), which allowed a 1 percent power uprate. Following the publication of the changes to 10 CFR 50, Appendix K, which allowed for an uncertainty less than 2 percent, Cameron submitted topical report ER-160P (ADAMS Accession No. ML010510372), a supplement to ER-80P. The NRC staff approved ER-160P by letter dated January 19, 2001 (ADAMS Accession No. ML010260074), for use in a power uprate of up to 1.4 percent. Subsequently, in an SE dated December 20, 2001 (ADAMS Accession No. ML013540256), the NRC staff approved ER-157P, Rev. 5 (ADAMS Accession No. ML013440078), for use in a power uprate of up to 1.7 percent using the CheckPlus system. The NRC staff also recently approved ER-157P, Rev. 8 and associated errata (ADAMS Accession Nos. ML081720323 and ML102950246). ER-157P, Rev. 8, corrects minor errors in Rev. 5, provides clarifying text, and incorporates revised analyses of coherent noise, non-fluid delays, and transducer replacement. It also adds two new appendices, Appendix C and Appendix D, which describe the assumptions and data that support the coherent noise and transducer replacement calculations, respectively.

McGuire 1 and 2 was originally designed with FW flow and temperature instrumentation consisting of ASME FW measurement nozzles, differential pressure transmitters, and thermocouples. Although the CheckPlus UFM system will be installed as part of the implementation of this LAR, 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 Cameron LEFM CheckPlus uses an ultrasonic 8-path transit time flowmeter. As discussed above, the CheckPlus design is described in Topical Reports ER-80P, ER-160P, and ER-157P that already have been approved by the NRC staff for generic use. The LEFM CheckPlus system will be used to develop a continuous calorimetric power calculation by providing FW mass flow and FW temperature input data to the plant computer system that is used for automated performance of the calorimetric power calculations.

The CheckPlus system consists of one flow element (spool piece) installed in each of the SG FW flow headers. The FW piping configurations are explicitly modeled as part of the CheckPlus meter factor and accuracy assessment testing performed at Alden Research Laboratories (ARL). The planned installation location of each CheckPlus conforms to the applicable requirements in Cameron's Installation and Commissioning Manual and Cameron topical reports ER-80P and ER-157P. The bounding uncertainty analysis is addressed in topical reports ER-823 and ER-874, which are included in a proprietary attachment to the LAR.

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

Early revisions of 10 CFR 50.46, and 10 CFR 50, Appendix K, required licensees to base their LOCA analyses on an assumed power level of at least 102 percent of the CLTP to account for power measurement uncertainty. The NRC later amended its regulation at 10 CFR 50, Appendix K, to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a power level higher than the previously licensed power. In the LAR, the licensee proposed to use a 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 0.3 percent. The licensee developed its LAR consistent with the guidelines in NRC RIS 2002-03.

3.1.1.2 Technical Evaluation

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

In Attachment 1 to RIS 2002-03, the NRC staff issued "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate [license amendment] Applications." This document provided guidance to licensees on one way to obtain NRC staff approval of their MUR LARs. In Section I of Attachment 1 to RIS 2002-03, the NRC staff provided guidance to licensees on how to address the issues of FW flow measurement technique and power measurement uncertainty in their MUR LARs. The following discusses the licensee's response to these guidelines in the LAR and the NRC staff's evaluation of these responses. Section I of Attachment 1 to RIS 2002-03 contains eight items for the licensee to respond to and each of these is discussed in turn.

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

Items A and B request the licensee to identify and reference the documents that form the regulatory basis for the LAR. The licensee provided this information in Section I.1.A and I.1.B of Enclosure 2 to the LAR. Item C requests "A discussion of the plant-specific implementation of the guidelines in the topical report and the [NRC] staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique." Section I.1.C of Enclosure 2 to the LAR provides the discussion of the plant-specific implementation of the applicable topical reports.

NRC Staff Conclusions on Items A, B, and C of Section I, Attachment 1 to RIS 2002-03

The NRC staff reviewed the licensee's response to items A, B, and C, and finds that the licensee has sufficiently addressed the plant-specific implementation of the Cameron LEFM CheckPlus System using the proper guidelines from the applicable topical reports. The NRC staff also evaluated this information against the regulatory requirements of 10 CFR 50, Appendix K, and found it to be acceptable.

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

The licensee's response to item D addresses the criteria established by the NRC staff in its approval of the FW flow measurement uncertainty technique used by the licensee in the LAR. When the NRC staff approved ER-80P and ER-157P, Revision (Rev.) 8, in NRC staff SEs dated March 8, 1999 and August 16, 2010, respectively, it established nine criteria (four criteria from ER-80P and five criteria from ER-157P) that licensees were to address in order to implement these topical reports at their facilities. The licensee addressed these criteria in Section I.1.D of Enclosure 2 to the LAR and in later supplements to the LAR that responded to NRC staff Request for Additional Information (RAI) questions. The NRC staff evaluated the licensee's approach to addressing each of these criteria.

The NRC staff evaluation for Criterion 1 from ER-157P, Rev. 8, is addressed Section 3.1.1.2.1.4 of this SE (i.e. the NRC staff's evaluation of the licensee's response to Items G and H of Section I, Attachment 1 to RIS 2002-03).

Criterion 1 from ER-80P

The licensee addressed Criterion 1 from ER-80P in Section I.1.D of Enclosure 2 to the LAR, which required a discussion of the maintenance and calibration procedures that will be implemented with the LAR.

The preventative maintenance program and continuous monitoring of the LEFM ensure that its performance remains bounded by the analysis and assumptions set forth by the vendor. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

The NRC staff reviewed the licensee's response to Criterion 1 from ER-80P and finds it acceptable, because the calibration and maintenance procedures (and associated documentation) will adequately ensure the incorporation of, and continued adherence to, the requirements set forth by the vendor.

Criterion 2 from ER-80P

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that this criterion does not apply to McGuire 1 and 2, as they do not have LEFMs installed at this time. McGuire 1 and 2 currently use ASME flow nozzles to measure FW flow to support the secondary calorimetric power measurements.

The NRC staff finds the licensee's response adequate to address Criterion 2.

Criterion 3 from ER-80P

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that

> The LEFM uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1, Instrument Society of America (ISA) Recommended Practice (RP) ISA RP 67.04 and Alden Research Laboratory Inc. calibration tests. This methodology has been used for instrument uncertainty calculations for multiple MUR power uprates and has been indirectly approved by the NRC in the acceptance of those uprates.
>
> The feedwater flow and temperature uncertainties are combined with other plant measurement uncertainties (steam temperature, steam pressure, feedwater pressure) to calculate the overall heat balance uncertainty as described in Section I.1.E below. This LEFM uncertainty calculation method is consistent with the current heat balance uncertainty calculation that uses the feedwater flow nozzles and [resistance temperature detectors] RTDs. The current calculation is based on a square-root-of-the-sum-of-the-squares (SRSS) calculation.

The FW flow and temperature uncertainties are combined with other plant measurement uncertainties (steam temperature, steam pressure, FW pressure) to calculate the overall heat balance uncertainty as described in the discussion of Item E in Section 3.1.1.2.1.3 below. These uncertainty calculations are based on an SRSS calculation. LEFM uncertainty calculations methods were provided in the plant-specific Cameron Engineering Reports ER-874 and ER-823. These calculations are consistent with Cameron Topical Reports ER-80P and ER-157P, which have been approved by the NRC staff. In addition, the licensee submitted Cameron reports ER-822 and ER-819, included as proprietary enclosures to the supplement to the LAR dated July 6, 2012, which summarized the calculations for the bounding analysis for thermal power uncertainty. ER-822 and ER-819 are consistent with the current heat balance uncertainty calculation that uses the FW flow nozzles and RTDs.

The licensee's calculations of the LEFM uncertainty arithmetically summed uncertainties for parameters that are not statistically independent and statistically combined with other parameters. The licensee combined random uncertainties using the SRSS approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of the Cameron LEFM CheckPlus System uncertainty, as described in the referenced topical reports, and is consistent with the guidelines in Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation" (ADAMS Legacy Accession No. ML993560062).

Furthermore, in Commitment 5 in Attachment 1 to the LAR, the licensee committed to perform acceptance testing following installation of the CheckPlus systems to ensure that the as-built parameters will be within the bounds of the error analyses. In Enclosure 6 to a supplement to the LAR dated July 6, 2012, the licensee committed to collect six months of data comparing the LEFM operating data with the venturi data to verify consistency between thermal power calculation based on LEFM and other plant parameters.

The NRC staff reviewed the Cameron topical reports described above and finds that the methodology used to calculate uncertainty is based on an accepted plant setpoint methodology and is consistent with the guidance in RG 1.105. The NRC staff, therefore, concludes that the licensee has adequately addressed Criterion 3 of ER-80P.

Criterion 4 from ER-80P

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that

> This criterion does not apply to McGuire, as the flow elements were tested and calibrated in a full-scale model of the McGuire Units 1 and 2 hydraulic geometry at the Alden Research Laboratory [ARL]. A bounding calibration factor for the McGuire Units 1 and 2 spool pieces was established by these tests and is included in the Cameron engineering reports for each unit. An Alden data report for these tests and a Cameron engineering report (ER-874 and ER-823 are included in Attachment 4 to this LAR) evaluating the test data have been prepared. A bounding uncertainty for the LEFM has been provided for use in the uncertainty calculation described in Section I.1.E below. A copy of the site-specific uncertainty analyses are provided in Attachment 4 to this License Amendment Request.

The NRC staff reviewed the licensee's statement that this criterion does not apply to McGuire 1 and 2 and, based on the information provided in the LAR and further supplements, the NRC staff concludes that the licensee addressed Criterion 4 of ER-80P.

In Enclosure 6 of a supplement to the LAR dated July 6, 2012, the licensee identified a licensee commitment to compare LEFM CheckPlus operating data to the venturi data to verify consistency between thermal power calculation based on LEFM and other plant parameters after final trial commissioning of the Cameron LEFM. The NRC staff reviewed this commitment and finds that it supports the NRC staff finding regarding the acceptability of the licensee's response to Criterion 4 of ER-80P.

Criterion 1 from ER-157P, Revision 8

The NRC staff evaluation for Criterion 1 from ER-157P, Rev. 8, is addressed in Section 3.1.1.2.1.4 of this SE (i.e. the NRC staff's evaluation of the licensee's response to Items G and H of Section I, Attachment 1 to RIS 2002-03).

Criterion 2 from ER-157P, Revision 8

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that

> McGuire Nuclear Station will not consider a CheckPlus system with a single failure as a separate category; this will be considered as an inoperable LEFM and the same actions identified in response to Criterion I from ER-1 57P, Rev. 8 above will be implemented.

The NRC staff has reviewed the licensee's statement and finds it acceptable.

Criterion 3 from ER-157P, Revision 8

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that

> As stated in response to Criterion 2 from ER-1 57P, Rev. 8 above, McGuire Nuclear Station will not consider a CheckPlus system with disabled components as a separate category; this will be considered as an inoperable LEFM and the same actions identified in response to Criterion 1 above will be implemented.

The NRC staff has reviewed the licensee's statement and finds it acceptable.

Criterion 4 from ER-157P, Revision 8

The licensee stated in Section I.1.D of Enclosure 2 to the LAR that

> The ASME feedwater measurement nozzles have a flow straightener immediately upstream. As discussed in Section 1.1.C above, the ASME feedwater measurement nozzles are located much greater than 4 [length/diameter] L/D from the planned location of the LEFMs. The planned location of the LEFMs is also upstream of the ASME feedwater measurement nozzles and will not include a flow straightener. Therefore, this criterion is not applicable to McGuire.

Operation with an upstream flow straightener is known to affect CheckPlus calibration to a greater extent than most other upstream hardware. If a licensee proposes this configuration, it must provide justification.

On August 24, 2009, while NRC staff members were at ARL, an effect of upstream tubular flow straighteners on CheckPlus calibration was discovered during ARL testing. This effect had not been documented and did not appear to apply to any previous CheckPlus installations. As a follow-up, additional tests were conducted with several flow straighteners and two different pipe / spool piece diameters to enhance the statistical data basis and to develop an understanding of the interaction between flow straighteners and the CheckPlus. The results are provided in the proprietary report ER-790, Rev. 1 (ADAMS Accession No. ML100840026).

Cameron concluded that two additional meter factor uncertainty elements are necessary if a CheckPlus is installed downstream of a tubular flow straightener and provided uncertainty values derived from the test results. The data also provide insights into the unique flow profile characteristics downstream of tubular flow straighteners and a qualitative understanding of why the flow profile perturbations may affect the CheckPlus calibration.

Cameron determined that the two uncertainty elements are uncorrelated and therefore combined them as the root sum squared to provide a quantitative uncertainty. The NRC staff reviewed the Cameron approach and judged it to be valid, but there was concern that the characteristics of existing tubular flow straighteners in power plants may not be adequately represented by samples tested in the laboratory. Therefore, any licensee that requests an MUR with the configuration discussed in this section should provide a justification for the claimed CheckPlus uncertainty that extends the justification provided in ER-790, Rev. 1.

The licensee has flow straighteners installed upstream of its ASME flow nozzles. The ASME flow nozzles are located more than 4 L/D in a horizontal run of main FW piping upstream from the planned LEFM location. The LEFMs will not have flow straighteners upstream of them and the flow straighteners located upstream of the ASME flow nozzles are a sufficient distance away that they will not affect the LEFM operation.

The NRC staff has reviewed the licensee's approach to evaluating and addressing the impact of upstream flow straighteners on CheckPlus calibration and has found that the licensee has acceptably addressed the effects of flow straighteners.

Criterion 5 from ER-157P, Revision 8

The licensee addressed the impact of steam moisture content on determining power measurement uncertainty in Section I.1.D of Enclosure 2 to the LAR.

The licensee specifically addressed the effects of moisture content in the steam generators at McGuire 1 and 2 for the Babcock & Wilcox International (BWI) Model CFR-80 steam generators installed in 1997. A test of moisture carry-over on a similar BWI Model CFR-80 steam generator at Catawba 1 in 1996 demonstrated a moisture content of 0.051 + 0.006 percent. Based on the test results for Catawba 1, the licensee conservatively assumed a moisture content uncertainty of 0.05 percent for McGuire 1 and 2. In its SE approving ER-157P dated August 16, 2010, the NRC staff stated:

> Some modern separators and dryers deliver steam with a moisture content in the 0.05 percent range, and these licensees often assume a zero moisture content that is conservative since the calculated power will be greater than actual power for such cases. No uncertainty is necessary, if there is no moisture.

The NRC staff considers that this uncertainty is small and not a significant factor in the calculation of the total power uncertainty of 0.29 percent. This is considered an insignificant factor because the total power uncertainty is calculated using the SRSS of all the independent uncertainty parameters and the contribution of this steam moisture is negligible to the total power uncertainty.

NRC Staff Conclusions on 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 (with the exception of Criterion 1 from ER-157P, Rev. 8, which is addressed in Section 3.1.1.2.1.4 of this SE as noted above). The licensee stated that Criterion 2 and 4 from the NRC staff's SE for ER-80P, and Criterion 2 and 3 from the NRC staff's SE for ER-157P, Rev. 8, were not applicable. The NRC staff reviewed these assessments by the licensee and found them acceptable. The NRC staff reviewed the licensee's evaluation of Criterion 1 and 3 from the NRC staff's SE for ER-80P, and Criterion 4 and 5 from the NRC staff's SE for ER-157, Rev. 8, and found them acceptable.

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

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

The licensee submitted Cameron Engineering Reports ER-819 and ER-822 in Enclosure 4 to the supplement to the LAR dated July 6, 2012. ER-822 and ER-819 summarize the bounding uncertainty analyses for thermal power determination using the LEFM CheckPlus System at McGuire 1 and 2, respectively. These two calculations provide analysis of the uncertainty contributions of the LEFM CheckPlus System to the overall RTP uncertainty of McGuire 1 and 2 in both its normal operation, as well as when operating in maintenance mode. These reports were prepared following the calibration of the spool pieces, when a precise estimate of the uncertainty in the profile factor became available. In addition, the as-built dimensions are input for all computations and the licensee ensured that the uncertainties in these dimensions lie within the bounding values used in the bounding analysis. Furthermore, in the LAR the licensee committed to perform commissioning tests for the LEFM CheckPlus System following installation at the plant which will ensure that the time measurement uncertainties are within the bounding values used in these reports.

In the LAR the licensee provided Table 1.1.E-1 which indicates that the uncertainty for the calorimetric inputs provided by the Cameron LEFM is 0.27 percent for McGuire 1 and 0.28 percent for McGuire 2. The LEFM thermal power uncertainty was combined with the non-LEFM uncertainties to obtain a bounding total power uncertainty of 0.29 percent for McGuire 1 and 2. These uncertainties were calculated utilizing the calculation methodology described in Cameron report ER-80P and ER-157P.

The NRC staff reviewed the calculations provided in ER-822 and ER-819 and determined that the licensee identified the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties and calculated the overall thermal power uncertainty in conformance with ER-157P, Rev. 8.

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty for McGuire 1 and 2, explicitly identifying parameters and their individual contribution to the power uncertainty. Therefore, the NRC staff concludes that the licensee has provided the information requested in Item E of Section I of Attachment 1 to RIS 2002-03.

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

Maintaining Calibration

The licensee responded that calibration of the LEFM will be ensured by preventive maintenance activities previously described in Item D, Criterion 1 from ER-80P, discussed above.

The NRC staff has evaluated the licensee's response and finds it acceptable.

Controlling Software and Hardware Configuration

The licensee described an approach to controlling software and hardware configuration in Section I.1.F.II of Enclosure 2 to the LAR.

The LEFM CheckPlus System is designed and manufactured per Cameron's quality assurance program (compliant with 10 CFR 50, Appendix B) and was procured according to the requirements of ANSI Standard 7-4.3.2-2003 and ASME NQA-1, 2008. Hardware configuration will be controlled in accordance with Duke Energy directive, NSD-301, "Engineering Change Program."

The LEFM software will be classified in accordance with Duke Energy directive EDM-801, "Cyber Security Risk Evaluation" and NSD-804, "Cyber Security for Digital Process Systems." Software will be classified, developed, tested, and controlled in accordance with NSD-806, "Digital System Quality Program." Implementation of the software will be performed under the design control process governed by EDM-601, "Engineering Change Manual."

Instruments that affect the power calorimetric, including the Cameron LEFM CheckPlus System inputs, are monitored by McGuire 1 and 2 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 the QA program, include instructions for notification of deficiencies and error reporting.

The NRC staff reviewed the licensee's approach to controlling software and hardware configuration and finds it acceptable.

Corrective actions and Deficiencies

In the LAR, Enclosure 2, Sections I.1.F.iii-v, the licensee identified its approach to performing corrective actions, reporting deficiencies to the manufacturer, and receiving and addressing manufacturer deficiency reports, respectively. The licensee indicated that it will monitor and perform corrective actions in accordance with its problem investigation program and work process manual. The licensee will also report deficiencies to the manufacturer in accordance with its problem investigation program and procurement specifications. The licensee will also receive and address manufacturer deficiency reports in accordance with the problem investigation program as well.

The NRC staff reviewed the licensees approach to addressing deficiencies and corrective actions and found it acceptable.

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

The licensee addressed a proposed allowed outage time (AOT), for the LEFM CheckPlus system, along with the technical basis for the AOT, in Section I.1.D of Enclosure 2 to the LAR (in the discussion of the licensee's response to Criterion 1 from the NRC staff's approval of ER-157, Rev. 8), and in supplements to the LAR dated July 6, 2012, and September 27, 2012. This discussion included a description of the proposed actions to reduce power level if the AOT is exceeded, as well as a discussion of the technical basis for the proposed reduced power level. This discussion was supplemented by letters dated July 6, 2012, and September 27, 2012, in response to NRC staff RAI questions regarding the licensee's approach to the AOT for the LEFM CheckPlus system. The supplement to the LAR dated September 27, 2012, changed the AOT proposed for the LEFM CheckPlus system from 7 days to 3 days and provided the basis for this change.

In its original submittal of the LAR, dated March 5, 2012, the licensee proposed an AOT of 7 days with a bounding uncertainty of 0.045 percent RTP upon loss of the LEFM signal. This AOT was determined by calculating the drift of a best estimate of reactor power, a weighted average of the secondary calorimetric power calculation to determine the plant power in the event of a loss of LEFM signal. For the purpose of calculating the drift of the secondary calorimetric parameter, the licensee performed a drift evaluation of one year's data, averaged at 10-minute intervals and reported every 15 minutes. The licensee's analysis was used to establish a bounding uncertainty of 0.045 percent RTP over a 7-day period for McGuire 1 and 2. In its supplement to the LAR dated July 6, 2012, the licensee responded to NRC staff RAI questions regarding the selection of a 7-day AOT and provided details of the drift evaluation it used to establish this AOT.

The NRC staff evaluated the licensee's RAI response dated July 6, 2012, and noted that previous MUR power uprate license amendment applications had received approval for only a 3-day (72-hours) AOT for similar conditions. The NRC staff also observed that an AOT of 3 days (72-hours) is consistent with Cameron's analysis and recommendations for operating with a failed LEFM. In discussions with the licensee, the NRC staff also expressed the position that the AOT for this condition should be based primarily on the time it takes to resolve the LEFM failure and not on the measurement of instrument drift. An AOT of 3 days for repair or replacement of inoperable instrumentation and control systems is an established safety practice in the nuclear power industry.

In a supplement to the LAR dated September 27, 2012, the licensee committed to implement a 3-day AOT for a non-functional LEFM System without application of the out-of-service allowance of 0.045 percent RTP. In its supplement to the LAR dated September 27, 2012, the licensee provided the following basis for their proposed 3-day AOT:

- When an LEFM System is non-functional, signals from an existing ASME flow nozzle will be used as input to the Secondary Calorimetric portion of the Rated Thermal Power (RTP) calculation in place of the LEFM System. During normal LEFM operations, the signals from the ASME flow nozzles are calibrated to the

LEFM signals and upon LEFM failure the ASME flow nozzle calibration is locked to the last good LEFM value.

- A statistical analysis and review of drift data for plant instrumentation which will provide the ASME flow nozzle signals to the Secondary Calorimetric portion of the RTP calculation demonstrates instrumentation and RTP drift should be insignificant over a 72-hour AOT period. This indicates that, without application of a bias based upon a bounding value of RTP [secondary calorimetric uncertainty] SCU, the McGuire Units can be operated for 72-hours without exceeding the licensed RTP limit when the ASME Flow Nozzle signals are used as an input to the Secondary Calorimetric portion of the RTP calculation in place of the LEFM System.

- A review of ASME Flow Nozzle fouling history demonstrates that fouling/de-fouling should not introduce significant error/drift over a 72-hour AOT period. This indicates that, without application of a bias based upon a bounding value of RTP SCU, the MNS Units [McGuire 1 and 2] can be operated for 72 hours without exceeding the licensed RTP limit when the ASME Flow Nozzle signals are used as an input to the Secondary Calorimetric portion of the RTP calculation in place of the LEFM System.

- It is expected that most issues rendering an LEFM System non-functional could be resolved within a 72-hour AOT.

- The NRC has approved a 72-hour AOT for previous MUR power uprate applications. Reference NRC to Shearon Harris correspondence dated May 30, 2012 ([ADAMS] Accession Number ML11356A096), NRC to Calvert Cliffs correspondence dated July 22, 2009 ([ADAMS] Accession Number ML091820366), and NRC to Limerick correspondence dated April 8, 2011 ([ADAMS] Accession Number ML110691095).

In its supplement to the LAR dated September 27, 2012, the licensee also provided the text of a Selected Licensee Commitment (SLC) which will be added to address functional requirements for the LEFMs and the appropriate Required Actions and Completion Times for when an LEFM is not functional. 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 3411 MWt (the licensed rated thermal power before approval of the LAR). The licensee stated that the SLC changes will be controlled using processes implemented to comply with 10 CFR 50.59.

The NRC staff reviewed the licensee's LAR and supplements to the LAR regarding its proposed AOT and concludes that the licensee has provided sufficient justification for the proposed 72-hour AOT and associated actions to reduce power level if the AOT is exceeded. Therefore, the NRC staff concludes that the licensee has provided the information requested by Items G and H of Section I of Attachment 1 to RIS 2002-03.

3.1.1.2.2 General Acceptance Criteria for UFMs

General acceptance criteria for UFMs apply to all aspects of testing in a certified facility, transfer from the test facility, initial operation, and long-term in-plant operation. These criteria are:

- Traceability to a recognized national standard. This requires no breaks in the chain of comparisons, all chain links must be addressed, and there can be no unverified assumptions.

- Calibration.

- Acceptable addressing of uncertainty, beginning with an initial estimate of the bounding uncertainty and continuing through all aspects of initial calibration in a certified test facility, transfer to the plant, initial operation, and long-term operation.

For CheckPlus, meeting these criteria includes documenting:

- Design and characteristics information,

- Calibration testing at a certified test facility,

- Any potential changes associated with differences between testing and plant operation including certification that initial operation in the plant is consistent with pre-plant characteristics predictions, and

- In-plant operation.

3.1.1.2.3 Initial Design and Characteristics

To determine volumetric flow rate, the Cameron CheckPlus UFM transmits an acoustic pulse along a selected path and records the arrival of the pulse at the receiver. Another pulse is transmitted in the opposite direction and the time for that pulse is recorded. Since the speed of an acoustic pulse will increase in the direction of flow and will decrease when transmitted against the flow, the difference in the upstream and downstream transit times for the acoustic pulse provides information on flow velocity. Once the difference in travel times is determined, the average velocity of the fluid along the acoustic path can be determined. Therefore, the difference in transit time is proportional to the average velocity of the fluid along the acoustic path.

The CheckPlus UFM provides an array of 16 ultrasonic transducers installed in a spool piece to determine average velocity in 8 paths. The transducers are arranged in fixtures such that they form parallel and precisely defined acoustic paths. The chordal placement is intended to provide an accurate numerical integration of the axial flow velocity along the chordal paths. Using Gaussian quadrature integration, the velocities measured along the acoustic paths are combined to determine the average volumetric flow rate through the flow meter cross section. Note that this process assumes a continuous velocity profile in the flow area perpendicular to the spool piece axis. Although the velocity profile can be distorted, the distortion cannot be such that the Gaussian quadrature process no longer provides an acceptable mathematical fit to the profile, such as may occur if the profile is distorted in a way that is not recognized by the CheckPlus due to an upstream flow straightener.

To obtain the actual average flow velocity, a calibration factor is applied to the integrated average flow velocity indicated by the UFM. The calibration factor for the CheckPlus UFMs is determined through meter testing at ARL and is equal to the true area averaged flow velocity divided by the flow velocity determined from the average meter paths to correlate the meter readings to the average velocity and, hence, to the average meter volumetric flow. The mass flow rate is found by multiplying the spool flow area by the average flow velocity and density. The mean fluid density is obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. Typically, the difference between an uncalibrated CheckPlus and ARL test results is less than 0.5 percent. This close agreement means that obtaining a correction factor for a CheckPlus UFM is relatively insensitive to error for operation under test conditions. Further, as discussed in this SE, correction factor is not a strong function of the difference between test and plant conditions and the same conclusion applies.

Use of a spool piece and chordal paths improves the dimensional uncertainties, including the time measurement of the ultrasonic signal, and enables the placement of the chordal paths at precise locations generally not possible with an externally mounted UFM. This allows a chordal UFM to integrate along off-diameter paths to more efficiently sample the flow cross section. In addition, a spool piece has the benefit that it can be directly calibrated in a flow facility, improving measurement uncertainty compared to externally mounted UFMs that were historically installed in nuclear power plant FW lines.

The NRC staff has reviewed the licensee's initial design and characteristics of the CheckPlus UFM and determined that the licensee acceptably addresses the aspects of UFM design discussed above in this section. Flow straighteners will not be used immediately upstream of the planned installations and other potential distortions of the flow profile are either absent or acceptably addressed in ARL testing.

3.1.1.2.4 Test Facility Considerations

Test facility considerations include test facility qualification, as well as test fidelity and range.

Test Facility Qualification

Calibration testing at a qualified test facility and at a nuclear power plant involves ensuring traceability to a national standard, understanding facility uncertainty, and facility operation. In the LAR, the licensee used Cameron reports that reference the work of ARL to provide traceability to National Institute of Standards and Technology (NIST) standards. The testing at ARL (ADAMS Accession No. ML072710557) was audited by the NRC staff in 2006 (ADAMS Accession No. ML060400418) and the NRC staff verified ARL's statement with respect to traceability to NIST standards. The NRC staff's audit found that ARL's processes and operation were consistent with the claimed facility uncertainties. The NRC staff also observed testing during a visit to ARL on August 24, 2009 (meeting notes at ADAMS Accession Nos. ML092680921 and ML092680922) and observed some improvements in test facility hardware. The NRC staff judged these changes would not change its previous conclusions regarding test operations and results. In ER-819 and ER-822, Cameron restated that "all elements of the lab measurements ...

are traceable to NIST standards." Consequently, the NRC staff finds that the references provide an acceptable basis for concluding that ARL meets the stated testing criteria.

Historically, all CheckPlus installations have been calibrated at ARL, including the McGuire 1 and 2 CheckPlus spool pieces. An NRC staff audit confirmed that ARL was providing acceptable test data for the configurations under test. Consequently, the NRC staff finds that the qualification of ARL with respect to CheckPlus testing is acceptable without further investigation or confirmation, provided test conditions remain consistent with the referenced conditions.

Test Fidelity and Test Range

Test fidelity, such as test versus planned plant configuration, test variations to address configuration differences, and potential effects of operation on flow profile and calibration, should be addressed on a plant-specific basis. In order for the NRC staff to complete its review, the LAR had to provide a comparison of the test and plant piping configurations with an evaluation of the effect of any differences that could affect the UFM calibration. Further, sufficient variations in test configurations must be tested to establish that test-to-plant differences have been bracketed in the determination of UFM calibration and uncertainty. Historically, calibration testing has acceptably covered upstream effects by applying a variation of configurations to distort the flow profile. Further, if the spool piece may be rotated during plant installation from the nominal test rotation, the effect of rotation should be addressed during testing.

In order for the NRC staff to complete its review, the LAR had to provide plant piping configuration drawings which must, at a minimum, include isometrics with dimensional information that describe piping, valves, FW flow meters, and any other components, from the FW pumps to at least 10 pipe diameters downstream of the FW flow meter that is most distant from the FW pump. Preferable are scale, three dimensional (3D) drawings in place of isometrics that show this information. Test information must include 3D drawings of the test configuration including dimensions.

ER-823 (ADAMS Accession No. ML12082A214) and ER-874 (ADAMS Accession No. ML12082A213) provide test configuration descriptions and drawings. The NRC staff reviewed the McGuire 1 and 2 pipe and instrumentation diagrams (P&IDs) that show the CheckPlus installation locations. The UFMs in loops A, B, C, and D will be installed upstream of the flow nozzles in the plants. The distances between the exit of the CheckPlus spool pieces and the downstream elbows in the tests are greater than six feet. As seen in the discussion of the "Evaluation of the Effect of Downstream Piping Configurations on Calibration" in Section 3.1.1.2.5.2 of this SE, this separation distance is large enough that there will be no effect on UFM calibration. The difference between the location of the downstream disturbance used in the calibration and that which exists in the plant has no impact on UFM uncertainty. For both McGuire 1 and McGuire 2 the hydraulic model configuration at ARL was designed to be a duplicate of the site configuration. All loops were tested with greater than 10 feet of straight pipe upstream of the UFM to the first non-straight pipe element, which is an elbow. Typically, weigh tank tests were run at different flow rates for each simulated FW loop. Tests included 100 percent and lower flow rates through the CheckPlus and some tests included an eccentric orifice upstream in the feedwater pipes containing the CheckPlus. Most test results were included in the reported main FW calculation. Tests included using eccentric orifices to restrict flow and induce swirl.

The NRC staff reviewed the test fidelity and test ranges used by the licensee. In the LAR, the licensee has included Cameron reports that acceptably address the test fidelity and range. The reports include test configurations as well as the variations in tests run. The NRC staff finds that the licensee has acceptably addressed potential differences in testing configuration compared to the potential installation configuration.

3.1.1.2.5 <u>In-Plant Installation and Operation of LEFMs</u>

In the LAR, the licensee address in-plant installation and operation of the CheckPlus LEFMs.

3.1.1.2.5.1 <u>Transfer from Test to Plant and In-Plant Installation</u>

For each LAR for a power uprate, the licensee must include an in-depth evaluation of the UFM following installation at its plant that considers any differences between the test and in-plant results. The licensee must also prepare a report that describes the results of the evaluation including such items as calibration traceability, potential loss of calibration, cross-checks with other plant parameters during operation to ensure consistency between thermal power calculation based upon the LEFM and other plant parameters, and final commissioning testing. The process used should be documented and a final commissioning test report should be available to the NRC staff for inspection.

Historically, the Check and CheckPlus UFMs are the only UFMs to have acceptably demonstrated UFM calibration traceability from the test facility to U.S. nuclear power plants. This traceability is possible due to the ability to provide the flow distribution / velocity profiles as a function of radius and angular position in the spool piece, the small calibration correction necessary to fit test data to UFM indication, and the demonstrated insensitivity to changes in operation associated with transfer changes and plant changes. Although other means have been used to measure flow rate, such as use of tracers in the FW, they have not attained the small uncertainty demonstrated by the CheckPlus LEFM.

Experience to date is that a UFM must provide flow profile information and calibration traceability when extrapolating from test flow rates and temperature conditions to plant conditions. Transfer uncertainty is associated with any changes in mechanical and operating conditions in the plant due to any installations or other modifications. Changes in mechanical conditions include mechanical perturbations due to such things as installation of a transducer, mechanical misalignment, and fidelity between the test and plant. Changes in operating conditions can arise from such things as noise due to pumps and valves, changes in flow profile (including swirl and flow rate), and temperature.

As discussed above, the test facility configuration and test parameters are expected to provide a basis for providing fidelity between the test and plant. However, an exact correspondence is not possible. Potential differences must be addressed during implementation of the UFM and licensees are expected to have the ability to both identify differences and address them during operation.

Installation of LEFMs at McGuire 1 and 2 is now in progress. The licensee addressed the uncertainties introduced by installation of LEFMs at McGuire 1 and 2 in ER-819 and ER-822, respectively. As discussed in Section 3.1.1.2.4 of this SE above, the NRC staff finds that the qualification of ARL with respect to CheckPlus testing is acceptable without further investigation or confirmation, provided test conditions remain consistent with the referenced conditions. ER-819 and ER-822 are referenced for transducer installation uncertainty. The content is essentially identical to that found in Appendix D of ER-157P-A, Rev. 8, which was approved by the NRC staff in an SE dated August 16, 2010. Consequently, the NRC staff finds that the licensee's treatment of transducer installation uncertainty is acceptable. The licensee showed that LEFM commissioning will include verification of ultrasonic signal quality and evaluation of actual plant hydraulic flow profiles as compared to those documented during the ARL testing. The commissioning tests for the Checkplus UFM to be performed at McGuire 1 and 2 will confirm that the as built uncertainties remain bounded by the testing analysis. The NRC staff has evaluated the licensee's approach to the commissioning test and finds it acceptable.

In addition, in a July 6, 2012, supplement to the LAR the licensee provided the following commitment:

> After the LEFM Checkplus system is installed and operational on the respective Unit, six months of data will be collected comparing the LEFM Checkplus operating data to the Venturi data to verify consistency between thermal power calculation based on LEFM and other plant parameters.
>
> The data will be available for NRC inspection seven months after the LEFM Checkplus system is installed and operational on the respective Unit.

The NRC staff has reviewed this commitment and finds it consistent with the approach the licensee has taken for transfer from test to plant and in-plant installation and finds it acceptable.

3.1.1.2.5.2 <u>In-Plant Operation</u>

Many of the calibration aspects associated with the transfer from a test facility to the plant apply during operation as valve positions change, different pumps are operated, and physical changes occur in the plant. The latter include such items as temperature changes, preheater alignment and characteristics changes, pipe erosion, pump wear, crud buildup and loss, and valve wear. Further, potential UFM changes, such as transducer degradation or failure, may also occur and the UFM should be capable of responding to such behavior. Either the UFM must remain within calibration and traceability must continue to exist during such changes, or the UFM must clearly identify that calibration and traceability are no longer within acceptable parameters. Past experience has shown that the CheckPlus has been capable of handling these operational aspects. Further, as stated above, UFM operation should be cross-checked with other plant parameters that are related to FW flow rate. Should such checking identify abnormal behavior, the validity of the final commissioning test report should be confirmed, and the final commissioning test report should be updated as necessary to reflect the new information. Further, the UFM must be considered inoperable if its calibration is no longer established to be within acceptable limits.

Section I.1 of Enclosure 2 to the LAR describes the training, calibration, maintenance, corrective action program, and procedures the licensee will use to ensure compliance with the requirements of 10 CFR 50, Appendix B. The NRC staff has evaluated Section I.1 of Enclosure 2 to the LAR and finds that the licensee's approach to in-plant operation is acceptable.

Operation with a failed LEFM CheckPlus system component was evaluated in Section 3.1.1.2.1.4 of this SE, which addressed items G and H of Section I, Attachment 1 to RIS 2002-03.

Spool Piece Dimensional Effects on UFM Response

Appendix A of ER-157P, Rev. 8, addresses the effect of variation in such spool piece dimensions as as-built internal diameter and sonic path lengths, path angles, and path spacings. The NRC staff has reviewed the licensee's approach for addressing these effects and finds it acceptable.

Transducer Installation Sensitivity

Transducers may be removed after ARL testing to avoid damage during shipping of the spool piece to the plant. Further, transducers may be replaced following failure or deterioration during operation. Replacement potentially introduces a change in position within the transducer housing that could affect the chordal acoustic path. Appendix D of ER-157P, Rev. 8, addresses replacement sensitivity by describing tests performed at the Caldon Ultrasonics flow loop. It also provides a comparison of test results to analyses for potential placement variations. This comparison shows that the test results are bounded by predicted behavior. One would expect an uncertainty associated with the test loop even if nothing was changed. This is not addressed in the ER-157P, Rev. 8, Appendix D. Rather, all of the test uncertainty is conservatively assumed to be due to transducer replacement. Further, the analyses predict a larger uncertainty than that obtained during testing, and the analysis uncertainty is used for transducer replacement uncertainty.

The NRC staff has evaluated this approach and judged it to be sufficiently conservative to cover the inability of the test loop to achieve flow rates comparable to those obtained in plant installations and to cover any analysis uncertainty associated with applications with pipe diameters that differ from the tests. Consequently, the NRC finds that transducer replacement has been acceptably addressed and that the ER-157P, Rev. 8, process for determining transducer replacement uncertainty is acceptable.

The Effects of Random and Coherent Noise of LEFM CheckPlus Systems

Appendix C of ER-157P, Rev. 8, provides a proprietary methodology for test- and plant-specific calculation of the contribution of noise to CheckPlus uncertainty. The NRC staff SE for this report dated August 16, 2010, has established that licensees may use this methodology in their MUR requests.

The LAR and ER-819 and ER-822 show that critical performance parameters, including signal-to-noise ratio, are continually monitored for every individual meter path. Alarm setpoints are established to ensure that the corresponding assumptions in the uncertainty analysis remain bounding. Signal noise will be minimized via strict adherence with Cameron design requirements.

In ER-823 and ER-874 the licensee reported test signal to ratios for random and coherent noise that were within specifications and that uncertainty attributable to the electronics and signal to noise ratio are included in the overall meter factor uncertainty.

The NRC staff has evaluated the test results and ER-819, ER-822, ER-823 and ER-874. The NRC staff finds that the licensee's approach for noise is sufficient to ensure that this topic is acceptably addressed.

Evaluation of the Effect of Downstream Piping Configurations on Calibration

Turbulent flow regimes exist when plants are near full power. This results in a limited upstream flow profile perturbation from downstream piping. Consequently, the effects of downstream equipment need not be considered for normal CheckPlus operation, provided that changes in downstream piping, such as the entrance to an elbow, are located greater than two pipe diameters downstream of the chordal paths. However, if the CheckPlus is operated with one or more transducers out of service, the acceptable separation distance is likely a function of transducer to elbow orientation. In such cases, if separation distance is less than five pipe diameters, it should be addressed.

As discussed in Section 3.1.1.2.4 of this SE above, separation from downstream components is needed so that CheckPlus operation will not be affected. The in-plant separation is greater than 4.75 feet to the nearest flow disturbance. Cameron's spool piece design guarantees distance between the acoustic paths and the next down stream flow disturbance. Cameron stated that the calibration will not be affected by the installation location at the plant and will not have an effect on CheckPlus operation.

The NRC staff has reviewed the licensee's approach to evaluation of the effect of downstream piping configurations on calibration and finds it acceptable.

3.1.1.3 NRC Staff Conclusions Regarding Power Measurement Uncertainty

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed LAR in support of implementation of an MUR power uprate. Based on the considerations discussed above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR.

The NRC staff has reviewed the licensee's response to RIS 2002-03, Attachment 1, Section I, and finds that the licensee has met the guidelines contained therein. The NRC staff finds that the licensee has adequately addressed the issues of FW flow measurement technique and power measurement uncertainty in its MUR LARs. The licensee has also adequately addressed general acceptance criteria for UFMs, adequately described the UFM design and characteristics, adequately addressed the test facility considerations, and adequately addressed issues with in-plant installation and operation of LEFMS.

3.1.2 Containment Systems

3.1.2.1 Regulatory Evaluation

For containment issues the regulation at 10 CFR, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4 (GDC 4), "Environmental and dynamic effects design basis," addresses the environmental qualification of systems, structures and components (SSCs) important to safety. The NRC staff reviewed the licensee's prediction of conditions in containment during postulated accidents.

No regulation specifically addresses the determination of the mass and energy (M&E) release into the containment following a postulated design basis accident (DBA). However, GDC 16, "Containment design," and GDC 50, "Containment design basis," address the requirements for the containment pressure resulting from the discharge of M&E into the containment as a result of a postulated design-basis LOCA.

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

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

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

The regulation at 10 CFR Part 50, Appendix J, Option B, defines Pa as "the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident as specified in the Technical Specifications." As discussed in the portion of Section 3.1.2.2 of this SE evaluating the "Short-Term and Long-Term LOCA Mass and Energy Release and Containment Analysis," the Pa values in the McGuire 1 and 2 TS Section 5.5.2, "Containment Leakage Rate Testing Program (CLRTP)," remain greater than the Pa values calculated for the LAR.

Review guidance in the area of containment safety analysis can be found in several sections of the Standard Review Plan (SRP) (ADAMS Package Accession No. ML070660036), including Section 6.2.1, "Containment Functional Design," (ADAMS Accession No. ML070220505), Section 6.2.1.1.B, "Ice Condenser Containments," (ADAMS Accession No. ML052340655), Section 6.2.1.2, "Subcompartment Analysis," (ADAMS Accession No. ML070620009), Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," (ADAMS Accession No. ML053560191), and Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures" (ADAMS Accession No. ML070620010).

3.1.2.2 <u>Technical Evaluation</u>

The NRC staff reviewed the following areas of containment performance analysis for the LAR: the short- and long-term LOCA peak containment pressure analysis, main steam line break

(MSLB) peak containment temperature analysis, MSLB with continued auxiliary feedwater (AFW) injection, long-term M&E data for LOCA, and long-term M&E data for MSLB.

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

With respect to the short-term and long-term LOCA M&E release and containment analysis, Section 6.2.1.3.1 of the McGuire 1 and 2 Updated Final Safety Analysis Report (UFSAR) discusses the short-term M&E data for a LOCA. McGuire 1 and 2 uses a Westinghouse Electric Company (Westinghouse) SATAN-V analysis. Section 6.2.1.2 of the McGuire 1 and 2 UFSAR discusses the containment subcompartment analysis. Both of these analyses are performed with the Westinghouse code TMD.

Section 6.2.1.1.3.1 of the McGuire 1 and 2 UFSAR discusses the containment short- and long-term peak pressure analysis following a LOCA. Section 6.2.1.3.2 of the UFSAR discusses the long-term M&E data for a LOCA.

As described in Section II.1.D, Item 43, "Containment Performance Analyses," of Enclosure 2 to the LAR, the McGuire 1 and 2 UFSAR LOCA analyses remain unaffected by the MUR power uprate since the M&E release input to the containment analysis was determined at 3479 MWt (102 percent of 3411 MWt). The NRC staff concurs that the current analysis of record (AOR) documented in the UFSAR is bounding. Therefore, the NRC staff finds that operation at the proposed power level is acceptable with respect to the LOCA containment analyses.

The NRC staff reviewed the LOCA response analysis and confirmed that the analysis was performed at 102 percent of 3411 MWt and remains bounding. Since the current analyses remain bounding, the Pa in TS Section 5.5.2, Leakage Rate Testing Program, remains unchanged for the MUR power uprate. Therefore, the NRC staff finds that TS 5.5.2 and the applicable McGuire 1 and 2 procedures developed to address 10 CFR 50, Appendix J, remain acceptable at MUR power uprate conditions.

MSLB M&E Release and MSLB with Continued AFW Injection Containment Response

With respect to the MSLB M&E Release and MSLB with Continued AFW Injection Containment Response, Section 6.2.1.1.3.3 of the McGuire 1 and 2 UFSAR discusses the MSLB peak containment temperature analysis, while MSLB with continued AFW injection is discussed in UFSAR Section 6.2.1.1.3.4. Section 6.2.1.4 of the UFSAR discusses the long-term M&E data for a MSLB.

The UFSAR MSLB analyses M&E release calculations assumed an initial power level of 102 percent at 3411 MWt (3479 MWt). This means that the UFSAR analyses bound the proposed case of a thermal power of 3469 MWt. The NRC staff, therefore, finds that operation at the proposed power level is acceptable with respect to the MSLB containment analyses.

Environmental Qualification

With respect to Environmental Qualification (EQ), the licensee states in part in Section II.1.D, Item 43 that:

> ... These analyses are performed to demonstrate peak containment pressures and temperatures are acceptable and to ensure the pressure and temperature profiles assumed in the Environmental Qualification (EQ) analyses are acceptable.

The NRC staff agrees that the current (i.e, UFSAR) analyses are bounding with respect to the LAR as discussed above. Therefore, the NRC staff finds that the EQ profile is conservative and acceptable with respect to operation at the proposed MUR power level described in the LAR.

Containment Systems

With respect to containment systems, the containment systems are provided to limit offsite releases following a DBA. These systems include the free standing steel containment, containment isolation system, ice condenser, containment spray, containment air return and hydrogen skimmer system, and annulus ventilation system. As indicated above, the existing containment analyses remain bounding. Therefore, these systems are not impacted by the MUR power uprate described in the LAR.

As the containment systems described in the LAR are not impacted by the MUR power uprate, the NRC staff has reviewed the licensee's evaluation of the containment systems described in the LAR and finds it acceptable.

Conclusion

The NRC staff finds that the current containment analyses remain bounding for the MUR power uprate described in the LAR. The current peak containment pressure is less than the containment design pressure and the EQ envelope remains bounding. In addition, the previously approved analytical methods remain acceptable. Therefore, the NRC staff, using the available SRP guidance, finds that GDC 4, GDC 16, GDC 38, and GDC 50 remain satisfied at MUR conditions described in the LAR.

3.1.3 Engineered Safety Features Heating, Ventilation and Air Conditioning Systems

Regulatory Evaluation

For Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning (HVAC) Systems, the NRC's regulations specify criteria for control room habitability and post-accident fission product control and removal. The NRC staff also used the guidance found in the SRP to guide its regulatory evaluation.

GDC 4, "Environmental and dynamic effects design bases," states in part that "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with ... postulated accidents, including loss-of-coolant accidents...." The effects of the release of post-accident fission products and toxic gases would be a consideration when evaluating McGuire 1 and 2 with respect to compliance with GDC 4.

GDC 19, "Control room," states in part that "... Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident…."

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

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

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

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

In its review, the NRC staff used specific criteria relevant to the evaluation of ESF HVAC Systems found in the SRP, Section 6.4, "Control Room Habitability System," (ADAMS Accession No. ML070550069), Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," (ADAMS Accession No. ML070190178), Section 9.4.1, Control Room Area Ventilation System," (ADAMS Accession No. ML070550045), Section 9.4.2, "Spent Fuel Pool Area Ventilation System," (ADAMS Accession No. ML070550038), Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System," (ADAMS Accession No. ML070550039), Section 9.4.4, "Turbine Area Ventilation System," (ADAMS Accession No. ML070550040), and Section 9.4.5, "Engineered Safety Feature Ventilation System," (ADAMS Accession No. ML070550041).

Technical Evaluation

The NRC staff reviewed the impact of the LAR on the control area ventilation system, the auxiliary building ventilation system, the diesel building ventilation system and the containment purge and ventilation system.

In Section VI.1.F, "Engineered safety features (ESF) heating, ventilation, and air conditioning systems," of Enclosure 2 to the LAR, the licensee states in part that the control area ventilation system, the auxiliary building ventilation system and the diesel building ventilation system "… remain bounded for the design basis (102 percent of 3411 MWt) for the MUR power uprate conditions. System design parameters are within the limits for all system components."

The licensee also stated in part in this section of the LAR that the containment purge and ventilation system "… is isolated and sealed during operation in Modes 1 through 4. The VP [containment purge and ventilation system] system is not put into operation until the unit is in Mode 5; therefore, the functions of the VP system are not affected by the 1.7 percent thermal power uprate."

The NRC staff reviewed the impact of the LAR on the control area ventilation system, the auxiliary building ventilation system, the diesel building ventilation system and the containment purge and ventilation system and, based on the information in the LAR, the NRC staff finds that the proposed increase in rated thermal power described in the LAR is acceptable with respect to the heating, ventilation, and air conditioning systems.

Conclusion

The NRC staff finds that the increase in heat loads in the control room and on the ESF ventilation systems is minimal and bounded by the current analyses. Therefore, McGuire 1 and 2 would continue to meet the criteria identified in GDC 4, GDC 19, GDC 41, GDC 60, GDC 61 and GDC 64. Applicable guidance in the SRP for evaluating the increase in heat loads in the control room and on the ESF ventilation systems is adequately addressed as well.

3.1.4 Plant Systems

3.1.4.1 Regulatory Evaluation

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

The NRC staff's review in the area of plant systems covers the impact of the LAR on the Nuclear Steam Supply System (NSSS) interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, and radioactive waste systems. The NRC staff's review is based on the guidance in the SRP Chapter 3 "Design of Structures, Components, Equipment, and Systems;" Chapter 6, "Engineered Safety Features;" Chapter 9, "Auxiliary Systems"; Chapter 10, "Steam and Power Conversion System;" and Chapter 11, "Radioactive Waste Management;" and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the LAR on the plant systems in Enclosure 2 of the LAR. The NRC staff reviewed the impact of the power uprate on major plant systems. The review covers the impact of the proposed MUR power uprate on:

- NSSS interface systems,
- containment systems,
- safety-related cooling water systems,
- SFP cooling analyses and systems,
- radioactive waste systems,
- ESF HVAC systems, and
- flooding analyses.

The review is conducted to verify that the licensee's analyses bound the proposed plant operation at the MUR power level described in the LAR, and that the results of licensee analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate.

3.1.4.2 Technical Evaluation

NSSS Interface systems

The NSSS interface systems include the main steam supply system, the condensate and FW system, and the AFW system.

The Main Steam Supply System (SM) is described in Section 10.3 of the UFSAR. The SM includes piping from the steam generators (SGs) to the main turbines, main FW pump turbines, AFW pump turbines, and moisture separator reheaters. The Main Steam Vent to Atmosphere (SV) and the Main Steam Vent to Condenser (SB) were included in the evaluation of main steam systems. The design bases of the SM, SB, and SV systems is to provide steam flow requirements at turbine inlet design conditions, dissipate heat from the reactor coolant system (RCS) following a turbine and/or reactor trip, provide main steam system overpressure protection, and provide steam to main FW and AFW pumps and other equipment. In the event of a main steam line rupture, the design basis of the SM, SB, and SV systems is to minimize positive reactivity effects and minimize containment temperature increase associated with a main steam line rupture within containment. In Section VI.1.A of Enclosure 2 to the LAR, the licensee states that "The review of the Main Steam System for the MUR uprate shows that all system functions will continue to be performed following the MUR uprate. The MUR power uprate conditions remain bounded by design as described in the McGuire UFSAR."

The Condensate and FW Systems are described in Section 10.4.7 of the UFSAR. Three motor driven hotwell pumps deliver condensate from the condenser hotwell through the condensate polishing demineralizers, the condensate coolers, the SG blowdown heat exchangers, and two stages of FW heating to the suction of the condensate booster pumps. Three motor driven condensate booster pumps deliver condensate through three stages of FW heating to the main FW pumps. Two steam turbine driven main FW pumps deliver FW through two high pressure heaters to a single FW distribution header, where FW is divided into four single lines to the SGs. The licensee completed a comparison between operating requirements for the 3469 MWt MUR power uprate condition and the 3411 MWt operating condition. The comparison demonstrated that the condensate and FW systems have sufficient design and operational margin to accommodate the MUR uprate. The licensee determined the MUR uprate conditions described in the LAR remain bounded by design as described in the UFSAR.

The AFW system provides FW to the SGs in the event of the loss of main FW. The AFW analysis is based on 102 percent of 3411 MWt, or 3479 MWt. The licensee stated that the analyzed core power level remains conservative and bounds the MUR power uprate (3469 MWt) described in the LAR. The licensee stated that AFW system maximum operating temperature and pressure remain essentially unchanged. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design or operation. There is no design change required for this system to operate at 3469 MWt. Therefore, the AFW system is capable of supporting the conditions identified in the LAR.

The NRC staff reviewed the information and evaluations performed by the licensee showing that the design of the NSSS interface systems at the increased power level is bounded by existing plant analyses, and, based on this information, finds them acceptable. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate because there is sufficient operating margin to produce an additional 1.7 percent RTP. The NRC staff concludes that an MUR power uprate will not challenge the NSSS interface systems. Therefore, the NRC staff finds that the NSSS interface systems are acceptable for the MUR power uprate conditions described in the LAR.

Safety-Related Cooling Water Systems

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

The NRC staff has reviewed the information provided in the LAR regarding the component cooling system and finds that the component cooling system is acceptable for the MUR power uprate conditions described in the LAR.

The nuclear service water system (NS) is described in Section 9.2.1 of the UFSAR. The NS provides assured cooling water for various Auxiliary Building and Reactor Building heat exchangers during all phases of station operation. Each unit has two redundant "essential headers" serving two trains of equipment necessary for safe shutdown, and a "non-essential header" serving equipment not required for safe shutdown. The licensee concluded that the MUR power uprate has no impact on the system or any of its major components and thus will have no impact on the system safety functions. The licensee determined that the existing design analysis bounds operation under MUR power uprate conditions described in the LAR.

The NRC staff has reviewed the licensee's analysis of the impact of the LAR on the NS and finds that the NS will perform acceptably upon implementation of the LAR.

The Ultimate Heat Sink (UHS) is described in Section 9.2.5 of the UFSAR. Two independent sources of nuclear service water are available to provide a normal supply of cooling water: Lake Norman via the Low Level Intake and the Standby Nuclear Service Water Pond (SNSWP). However, to dissipate the decay heat rejected during a unit LOCA plus a unit cooldown, the SNSWP is the only source qualified as the UHS. The licensing basis thermal analysis of the SNSWP assumes an initial condition of 102 percent of RTP (3479 MWt) for both units. This bounds the conditions after implementation of the LAR.

The NRC staff has reviewed the information above and finds that the UHS will perform acceptably upon implementation of the LAR.

The NRC staff has reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the analyses that show that these systems were evaluated for 102 percent RTP, the

NRC staff finds that there is reasonable assurance that the systems will perform acceptably after implementation of the LAR.

SFP Storage and Cooling

The principal function of the SFP storage and cooling system is to provide storage and cooling of the spent fuel. Section 9.1.3.1.1 of the UFSAR states that "The existing Spent Fuel Cooling System is designed to maintain the spent fuel pool water temperature within acceptable limits under normal and maximum heat load conditions...." The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The licensee stated the current analysis for SFP heat loads was performed at 3479 MWt (i.e., 102 percent of original thermal power of 3411 MWt).

The NRC staff does not expect that implementation of the LAR will result in a significant change to the operation of the SFP storage and cooling system. The NRC staff, therefore, has reviewed the licensee's conclusion and finds that the SFP storage and cooling system will not be impacted by implementation of the LAR.

Radioactive Waste Management Systems

The Radioactive Waste Management Systems: Waste Gas (WG); Liquid Waste Recycle (WL); Liquid Waste Monitor and Disposal (WM); are described in Section 11 of the UFSAR. The licensee evaluated the liquid and gaseous radioactive waste systems (WG, WL, and WM) for operation at the increased power level described in the LAR. The licensee stated in the LAR that:

> The WL and WM systems have no direct interface with the power cycle, and therefore, the MUR will have no impact on these systems' ability to fulfill their functions. These systems are also credited for performing containment isolation for mitigating design basis events, which were analyzed at 102 percent [of RTP, i.e., 3479 MWt]. Therefore, the WL and WM systems are not impacted by the MUR.

Based upon the information and evaluations performed by the licensee to show that the design of the liquid and gaseous radioactive waste systems at the increased power level is bounded by existing plant analyses, the NRC staff finds that the liquid and gaseous radioactive waste systems would perform acceptably after implementation of the LAR.

The Nuclear Solid Waste Disposal (WS) System is designed to contain solid radioactive waste materials as they are produced in the station, and to provide for their storage and preparation for eventual shipment to an appropriate disposal facility. The licensee stated "the WS system has no direct interface with the power cycle, and therefore, implementation of the LAR will have no impact on this system."

The NRC staff has evaluated the licensee's analyses of the WS system and finds that the WS would perform acceptably after implementation of the LAR.

Flooding analysis

Internal flooding of the turbine building, auxiliary building, diesel generator rooms, and the main steam dog house are addressed in Sections 3.6, 6.3, 6.5, 7.6, 8.3, 9.2, 9.3, 9.5, and 10.4 of the UFSAR. The licensee completed an engineering evaluation of the potential impact of the LAR on internal flooding in the building and rooms currently discussed in the UFSAR, as well as inside containment and in the annulus. No significant increases in fluid volumes in storage tanks or maximum flow rates through fluid system piping were identified. The licensee, therefore, determined that the existing flood analysis remain valid and are not affected by operating at the increased power level described in the LAR.

Based upon the information and evaluations performed by the licensee to show that the effects on internal flooding at the increased power level described in the LAR are bounded by existing plant analyses, the NRC staff finds that the internal flooding analysis contained in the LAR is acceptable.

High Energy Line Break

The licensee evaluated the consequences of a high energy line break (HELB) inside the containment building and the turbine building with respect to impact on safety-related equipment. High-energy pipe breaks are analyzed for piping for which the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200 °F. High-energy pipe cracks are postulated in piping for which either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200 °F. The licensee's evaluation concluded that no new postulated line break locations would be introduced by the increase in power level described in the LAR. In addition, no existing segments classified as non-high energy would become high energy. No new lines are added, no break locations changed, and no change is made to the assumed blowdown from any postulated break. There is, therefore, no impact on the HELB analysis that was originally performed for McGuire 1 and 2.

Based upon the information and evaluations performed by the licensee to show that the effects from a HELB at the increased power level described in the LAR is bounded by existing plant analyses, the NRC staff finds that the 1.7 percent increase in power acceptable in regards to HELBs.

3.1.4.3 Conclusion for Plant Systems Impacts

The NRC staff has reviewed the licensee's safety analyses of the impact of implementation of the LAR on the major plant systems. The NRC staff concludes that the results of licensee's analyses related to these areas would continue to meet the applicable acceptance criteria following implementation of the LAR. Therefore, the NRC staff concludes that the impact of changes to plant systems described in the LAR and the proposed changes to the technical specifications will be acceptable.

3.1.5 Accident Analyses

In the LAR, the licensee generally concluded that existing analyses bounded the uprated plant operation conditions with reduced uncertainty. The analyses were shown to be bounding in one of three different ways:

- For analyses that assume steady-state plant operation with a core power of 3479 MWt, there is a 2 percent margin for power measurement uncertainty at the CLTP (3411 MWt). These analyses are bounding also of plant operation at the MUR RTP of 3469 MWt, with an operating margin of 0.3 percent, which is equal to the stated 0.3 percent calorimetric power measurement uncertainty. For analyses that assume a steady-state plant operation with a core power of 3469 MWt, there is a 1.7 percent margin for power measurement uncertainty at the CLTP, 3411 MWt. This analysis is used for DNB and assumes DNB does not occur. The method for this analysis is explained below.

- For analyses that assume steady-state plant operation with a core power of 3411 MWt, the licensee evaluated accident or transient, and reanalyzed as necessary.

- Zero-power transients were not reanalyzed.

- A summary of the licensing basis transients and accidents is contained in Table 3-1 of this section of the SE.

In the LAR, the licensee used Statistical Core Design (SCD) methodology report DPC-NE-2005-PA, "Duke Energy Carolinas Thermal-Hydraulic Statistical Core Design Methodology." SCD statistically combines the uncertainty in core power, RCS flow, RCS temperature, and RCS pressure with the uncertainties in nuclear peaking to produce a statistical departure from nucleate boiling ratio (DNBR) limit. The analysis is performed to ensure that no DNB occurs. The SCD DNBR limit uses a thermal power uncertainty of ±2 percent RTP. This is greater than the proposed MUR uncertainty of ±0.3 percent RTP. The analysis is performed at 101.7 percent CLTP and includes the RTP uncertainty. The NRC staff concluded that this approved method bounds the proposed MUR uprate conditions.

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 utilized such an approach in its review of the LAR. The NRC staff did not conduct a detailed review of the licensee's analyses that were performed at 102 percent of the CLTP or the licensee's DNB analyses that were performed at 101.7 percent. For these analyses, the NRC staff found that existing AORs bound plant operation. Thus, the NRC staff finds that these AORs are acceptable without a detailed review.

Table 3-1 of this SE summarizes those areas for which the existing AORs do not bound the plant operation described in the LAR. These are analyses that received a detailed NRC staff review, consistent with the guidance of RIS 2002-03. The following sections of the SE will describe the NRC staff review of those areas.

The licensee found that the LOCAs needed to be reanalyzed prior to implementation of the MUR. This is discussed later in this section of the SE. The licensee found that there are no other accidents or transients where the existing AORs do not bound plant operation at the proposed uprated power level.

LOCAs

The licensee found that the current LOCA analysis was not bounding for the proposed MUR power level. Because of this, the licensee will re-evaluate the LOCAs prior to implementation of the MUR. The analysis of the LOCAs can be found in Section 15.6.5 of the McGuire 1 and 2 UFSAR. The current best-estimate large-break LOCA analysis assumes 101 percent of CLTP plus 1 percent uncertainty. The licensee has determined that this peak clad temperature (PCT) does not bound the power uprate conditions. The licensee will include a new best-estimate large-break LOCA analysis performed at 101.7 percent power with a 0.3 percent uncertainty. This will result in a 16 °F increase in PCT which will still be below the PCT limit.

The current small-break LOCA analysis starts from 3479 MWt and bounds the proposed uprated power level of 3469 MWt plus 0.3 percent uncertainty.

The licensee provided a commitment to reanalyze the LOCAs. The licensee also stated that the five criteria of 10 CFR 50.46 will continue to be met following a LOCA from the MUR uprate power level.

Conclusion for Accident Analyses

The NRC staff reviewed the current AORs. Most of the current AORs are based on operation at 3479 MWt which is 102 percent CLTP (3469 MWt for DNB considerations which is 101.7 percent CLTP). The LAR is based on the use of a Cameron LEFM Check Plus system that would decrease the uncertainty in the FW flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.3 percent. In these cases, the proposed MUR rated thermal power of 3469 MWt is bounded by the current AORs and the NRC staff finds them acceptable without performing a detailed review, consistent with the guidance contained in RIS 2002-03.

The NRC performed a detailed review of the licensee's LOCA analyses. The licensee found that the current LOCA analysis was not bounding for the proposed MUR power level and provided a commitment to reanalyze the LOCAs. The licensee also stated that the five criteria of 10 CFR 50.46 will continue to be met following a LOCA from the MUR uprate power level. The NRC staff reviewed the licensee's approach to reanalysis of the LOCAs and finds it acceptable.

Table 3-1 – Evaluation of Accident and Transient Analyses

Transient/Accident	Analytic Power Level (percent CLTP)	Review Comments
FW System Malfunction Causing an Increase in FW Flow	101.7	Acceptable
Excessive Increase in Secondary Steam Flow	101.7	Acceptable
Inadvertent Opening of a SG Relief or Safety Valve	0	Acceptable
Steam System Piping Failure	0	Acceptable
Turbine Trip	102	Acceptable
Loss of Non-Emergency alternating current (AC) Power to the Station Auxiliaries	102	Acceptable
Loss of Normal FW Flow	102 and 101.7	Acceptable
FW System Pipe Break	102 and 101.7	Acceptable
Partial Loss of Forced Reactor Coolant Flow	101.7	Acceptable
Complete Loss of Forced Reactor Coolant Flow	101.7	Acceptable
Reactor Coolant Pump (RCP) Shaft Seizure	102 and 101.7	Acceptable
Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal From a Subcritical or Low Power Startup Condition	0	Acceptable
Uncontrolled RCCA Bank Withdrawal at Power	10.17 and 101.7	Acceptable
RCCA Misoperation	101.7	Acceptable
Startup of an Inactive RCP at an Incorrect Temperature	50.85	Acceptable
Spectrum of RCCA Ejection Accidents	102	Acceptable
Inadvertent Operation of ECCS During Power Operation	0	Acceptable
Inadvertent Opening of a Pressurizer Safety or Relief Valve	101.7	Acceptable
Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment	100	Acceptable
Steam Generator Tube Failure	101.7	Acceptable
LOCAs	102	See discussion above

Fuel Handling Accidents in the Containment and Spent Fuel Storage Buildings	102	Acceptable
Anticipated Transients Without Scram	102	Acceptable

3.2 Engineering and Materials

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

The NRC staff's review in the area of reactor vessel (RV) integrity focuses on the potential impact of the LAR on pressurized thermal shock (PTS) calculations, neutron fluence calculations, RV pressure-temperature (P-T) limits, upper shelf energy (USE) evaluations, the RV surveillance capsule withdrawal schedules, and the integrity of RV internals. The NRC staff review was conducted in accordance with the guidance contained in RIS 2002-03 to verify that, following implementation of the LAR, the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

3.2.1.1 Pressurized Thermal Shock (PTS)

Regulatory Evaluation

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

Technical Evaluation

The licensee provided its PTS evaluation in Section IV.1.C.i of Enclosure 2 of the LAR which states in part that

> PTS screening calculations were performed for the McGuire Units 1 and 2 reactor vessel beltline materials using the 60-year end-of-life extension (EOLE) neutron fluence values. ... McGuire Units 1 and 2 reactor vessel beltline materials will continue to meet the 10 CFR 50.61 PTS screening criteria (270 °F for plates, forgings, and axial welds, and 300 °F for circumferential welds). For McGuire Unit 1, the limiting RT_{PTS} value is 203 °F, which corresponds to Lower Shell Longitudinal Welds 3-442A,B,C (Heat # 21935/12008), using credible Diablo Canyon [Nuclear Power Plant] Unit 2 surveillance data. For McGuire Unit 2, the limiting RT_{PTS} value is 148 °F, which corresponds to Lower Shell Forging 04.

The NRC staff found that the limiting RT_{PTS} value for McGuire 1 described in the initial LAR is not consistent with the value reported in its license renewal application (LRA). The NRC staff evaluated the licensee's LRA in NUREG-1772, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2," (ADAMS Package No. ML030850251). Section 4.2.2.2 of NUREG-1772 states that

> ... Using a limiting fluence of 2.73×10^{19} n/cm^2 at EOLE, the applicant's revised PTS assessment projected the RT_{PTS} values for these welds [the McGuire 1 limiting weld] to be 253 °F using all relevant surveillance capsule data for the heat No. 21935/12008, as obtained from docketed information from the Diablo Canyon 2 RV material surveillance program (inclusive of fracture toughness tests performed on test specimens from Diablo [Canyon] 2 capsules U, X, Y, and V)....

The NRC staff evaluated the fluence of 2.13×10^{19} n/cm^2 described in the LAR and adjusted the LRA RT_{PTS} value from 253 °F to 238 °F, but this value is still far greater than the limiting RT_{PTS} value of 203 °F described in the LAR. Therefore, the NRC staff issued RAI question 16 by letter dated May 22, 2012.

The licensee responded to RAI question 16 by a supplement to the LAR dated June 21, 2012, and indicated that the Diablo Canyon 2 surveillance capsule data was reevaluated in December 21, 2011, as documented in WCAP-17315-NP, "Diablo Canyon, Units 1 and 2, Pressurized Thermal Shock and Upper-Shelf Energy Evaluations" (ADAMS Accession No. ML12009A070). WCAP-17315-NP updated the surveillance capsule fluence values at the clad/base metal interface in accordance with WCAP-14040-A, Rev. 4, "Methodology Used to Developed Cold Overpressure Mitigating System Setpoints and RCS [Reactor Coolant System] Heatup and Cooldown Limit Curves," May 2004 (ADAMS Accession No. ML050120209).

The NRC staff reviewed the detailed calculations in Tables 1 through 4 of Enclosure 2 to the licensee's letter dated June 21, 2012, focusing on the impact on the McGuire 1 limiting weld RT_{PTS} value due to the updated fluence value, the inclusion of the temperature adjustment, and the credibility conclusion. The NRC staff concluded that the new information included in the supplement to the LAR, especially the updated credibility conclusion, which reduced the margin term from 56 °F to 28 °F, successfully explained the discrepancy. Hence, the limiting RT_{PTS} value of 203 °F described in the LAR is valid and does not exceed the PTS screening criteria. The first part of RAI question 16 is resolved.

The second part of RAI question 16 requests that the licensee confirm for McGuire 1 that (1) the peak fluence value for the intermediate shell plate longitudinal welds 2-442A, 2-442B, and 2-442C has been used for all three welds to simplify the classification of these welds in the LAR, and that (2) the similar simplification has also been applied to the lower shell plate longitudinal welds 3-442A, 3-442B, and 3-442C in the LAR. The licensee's response to RAI question 16 in its supplement to the LAR dated June 21, 2012, confirmed this. Therefore, the NRC staff finds that issues related to RAI question 16 are resolved.

Based on the above evaluation, the NRC staff accepts the licensee's limiting RT_{PTS} value of 203 °F for McGuire 1. The NRC staff also verified that the licensee's RT_{PTS} information for other McGuire 1 RV materials and for all McGuire 2 RV materials, except for fluence value-related data, is consistent with that in the LRA. Therefore, the NRC staff also accepts the licensee's limiting RT_{PTS} value of 148 °F for McGuire 2.

Conclusion

Since the RT_{PTS} values for the limiting RV beltline materials of McGuire 1 and 2 are lower than the PTS screening criterion of 270 °F for the RV axial welds and forgings, the NRC staff concludes that after implementation of the LAR, the McGuire 1 and 2 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 (P-T) Limits and Use

Regulatory Evaluation

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

Technical Evaluation

The licensee provided its P-T limit evaluation in Section IV.1.C.iii of Enclosure 2 of the LAR and its USE evaluation in Section IV.1.C.v. The NRC staff found that the current TS P-T Limits and low temperature overpressure protection system (LTOPS) setpoints for McGuire 1 are based on one quarter or three quarters of the RV wall thickness (¼T or ¾T) adjusted reference temperature (ART) values of 202 °F and 146 °F for the limiting material - the lower shell longitudinal welds. For McGuire 2, the limiting material is the lower shell forging 04, and the corresponding ¼T and ¾T ART values are 123 °F and 91 °F.

In the LAR, resolution of the discrepancy that was raised in RAI question 16, regarding the documented McGuire 1 LRA RT_{PTS} value and the LAR RT_{PTS} value for 54 EFPY affected the NRC staff's acceptance of the limiting ART value for 34 EFPY for the LAR P-T limit evaluation for McGuire 1. As indicated in the NRC staff's evaluation of the licensee's response to RAI question 16 in Section 3.2.1.1 of this SE, the NRC staff concluded that RAI question 16 is resolved. Therefore, the licensee's limiting ART values of 155 °F (¼T) and 110 °F (¾T) for 34 EFPY for the McGuire 1 P-T limits in the LAR are valid and acceptable.

Similar to the PTS evaluation for McGuire 2, after considering the difference in fluence values for the P-T limits in the current TSs and the LAR, the NRC staff found that the licensee's limiting ART values of 120 °F (¼T) and 87 °F (¾T) for 34 EFPY for McGuire 2 in the LAR are consistent with those in the current TSs and are, therefore, acceptable. The licensee summarized these ART values for the current TS and the LAR in Table IV.1.C-9 of Enclosure 2 to the LAR for McGuire 1 and in Table IV.1.C-10 for McGuire 2. Both tables indicated that the current TS ART values are bounding. However, to ensure that both the much higher stresses associated with RV discontinuities, as well as the lowest operating temperature requirement of the RCPB, are considered in the P-T limits, the NRC staff developed a generic RAI question for all relevant LARs, power uprates, and LRAs which contain P-T limit evaluations. This generic RAI question was raised during the NRC staff review of the LAR and the NRC staff sent RAI question 41 by letter dated August 1, 2012, which states that

> 10 CFR Part 50, Appendix G, Paragraph IV.A states that, "the pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth in [paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]..." Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the RV beltline (neutron fluence $\geqslant 1 \times 10^{17}$ n/cm^2, E > 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence < 1×10^{17} n/cm^2, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the ASME Code, Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all RCPB components specified in Section III, NB-2332(b).
>
> The P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials due to the following factors:
>
> 1. RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.
>
> 2. Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P -T limits than those for the RV beltline shell materials.
>
> Consequently, please describe how the current P-T limit curves at 34 EFPY for McGuire, Units 1 and 2 and the methodology used to develop these curves, considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G in the MUR power uprate application.

In its supplement to the LAR dated August 15, 2012, the licensee responded to RAI question 41 and agreed that a license condition would be used to address RAI question 41. In its supplement to the LAR dated March 7, 2013, the licensee provided the text of the license condition to address NRC staff RAI question 41.

The NRC staff reviewed the licensee's supplements to the LAR dated August 15, 2012, and March 7, 2013. Since this RAI question is related to a generic RAI question being asked during the review of all applications for license renewal, power uprate, P-T limit changes, or pressure temperature limit report changes, it is also being tracked as a separate issue. As part of its review of this LAR, the NRC staff determined that a license condition regarding P-T limits was needed in order to find that the changes identified in the LAR would not impact the safety margins required for the necessary structural integrity assessments.

The implementation date of one year from approval of the LAR is appropriate because the P-T limits are based on a postulated flaw of ¼T of the RV thickness. In its supplement to the LAR dated September 27, 2012, the licensee responded to the NRC staff's RAI question regarding the inservice inspection records and indicated that no flaw was identified in the McGuire 1 RV, but noted that two flaws were identified in the McGuire 2 RV: a flaw confined in the cladding and a subsurface flaw of a depth of 0.5 inch. Since these flaws are much smaller than the ¼T flaw postulated for the P-T limits, no immediate safety concern exists. Further, the supplement to the LAR dated September 27, 2012, revealed that there is approximately an additional 10 EFPY's operation before expiration of the current 34 EFPY P-T limits, indicating additional margin in the current P-T limits.

With the licensee's agreement to the license condition regarding NRC staff RAI question 41 as documented in its supplement to the LAR dated March 7, 2013, the NRC staff accepts the licensee's P-T limit evaluation.

Regarding the USE evaluation, Section IV.1.C.v of Enclosure 2 to the LAR stated that "The projected EOLE Charpy USE decreases due to MUR power uprate fluence at the ¼T location were calculated per Regulatory Guide 1.99, Rev. 2...." It further stated that "For McGuire Unit 1, the limiting projected [1/4T] USE value is 60.5 ft-lbs, which corresponds to Intermediate Shell Longitudinal Welds 2-442A,B,C (Heat # 20291/12008), using surveillance data. For McGuire Unit 2, the limiting projected USE value is 61.8 ft-lbs, which corresponds to Bottom Head Ring 03."

The limiting USE values reported in the LRA are 53 ft-lbs for the nozzle shell plate B5011-2 for McGuire 1 and 55 ft-lbs for nozzle shell to intermediate shell weld for McGuire 2. To resolve these discrepancies, the NRC staff issued RAI question 17.

In its supplement to the LAR dated June 21, 2012, responding to NRC staff RAI question 17, the licensee's stated that

> The USE evaluations in the [LRA] utilized the peak vessel fluence for all of the three shell course plates (nozzle, intermediate and lower).... In the MUR LAR, the material specific fluence value was used for the nozzle shell plates. The projected surface fluence on the nozzle shell plates is 0.0547×10^{19} n/cm^2, which resulted in the ¼T fluence value of 0.033×10^{19} n/cm^2 at 54 EFPY.

Based on the above, the NRC staff concluded that the nozzle shell plate B5011-2 is no longer the limiting USE material for McGuire 1 in the LAR because its fluence at the ¼T location has been revised from 1.83 x 10^{19} n/cm^2 (in the LRA) to 0.033 x 10^{19} n/cm^2 (in the LAR). The NRC staff has confirmed these values, and, therefore, the first part of RAI question 17 regarding McGuire 1 is resolved.

For the discrepancy associated with the intermediate shell longitudinal welds 2-442A, 2-442B, and 2-442C of McGuire 1, the licensee's June 21, 2012, response to RAI question 17 clarified that, after considering all RV surveillance data, the resulting USE percent decrease was higher (46 percent) than the values documented in the McGuire 1 LRA (36 percent), which resulted in the EOLE USE dropping from 72 ft-lbs to 60.5 ft-lbs. The NRC staff has confirmed these values, and therefore, the second part of RAI question 17 regarding McGuire 1 is resolved.

For the question regarding the fact that the nozzle shell to intermediate shell weld is no longer the limiting USE material for McGuire 2 in the LAR, the licensee's June 21, 2012, response to RAI question 17 clarified that, instead of the peak fluence for the RV, the material-specific fluence was used for this weld. The NRC staff has reviewed this approach and finds it acceptable. Regarding the initial USE of ">71" ft-lbs in Table IV.1.C-12 of Enclosure 2 to the LAR, the licensee's June 21, 2012, response to RAI question 17 clarified that the initial USE is based on the highest energy at 10 °F from the Certified Materials Test Report. The NRC staff considers this approach conservative and acceptable. The NRC staff has also reviewed and found acceptable the licensee's explanation regarding the bottom head ring 03 with an initial USE of ">71" ft-lbs. Hence, the part of RAI question 17 regarding McGuire 2 is resolved.

The NRC staff performed independent calculations for the rest of the RV beltline materials and found only minor discrepancies between the EOLE USEs provided in the LAR and the values obtained by the NRC staff. In summary, the NRC staff has found that the USEs calculated for the LAR are above 50 ft-lbs as required by 10 CFR Part 50, Appendix G.

Conclusion

The licensee addressed the impact of the LAR on the McGuire 1 and 2 USE evaluations. These analyses are documented in Enclosure 2 to the LAR, as supplemented by the licensee's letter dated June 21, 2012, responding to the NRC staff's RAI questions. Since the EOLE USEs for the RV materials used at McGuire 1 and 2 are above 50 ft-lbs, the NRC staff concludes that the RV beltline materials for McGuire 1 and 2 will continue to satisfy the USE criteria specified in 10 CFR Part 50, Appendix G, upon implementation of the LAR.

For the P-T limit evaluation, the NRC staff concludes that the RV beltline materials for McGuire 1 and 2 will continue to satisfy the P-T limit requirements specified in 10 CFR Part 50, Appendix G, upon implementation of the LAR. For the non-beltline RV material and the RCPB material, the NRC staff reviewed the licensee's evaluation and its acceptance of a license condition to address concerns regarding the P-T limits for all beltline and non-beltline RV materials, as well as the lowest service temperature of all ferritic RCPB. With an implementation date of one year from approval of the LAR for the license condition, the NRC staff finds the licensee's approach acceptable.

3.2.1.3 RV Material Surveillance Program

Regulatory Evaluation

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. The regulation at 10 CFR Part 50, Appendix H, identifies the requirements for the design and implementation of the RV material surveillance program.

Technical Evaluation

The licensee provided its evaluation of the RV material surveillance program in Section IV.1.C.vi of Enclosure 2 of the LAR. This Section states,

> The five required in-vessel surveillance capsules have been withdrawn and tested to date for McGuire Unit 1.... The four required in-vessel surveillance capsules have been withdrawn and tested to date for McGuire Unit 2. The remaining capsules for both units have also been withdrawn, but the specimens have not been tested. The specimens are stored for potential future use. Since all of the surveillance capsules have been withdrawn from the McGuire Units 1 and 2 reactor vessels, there is no longer a need to recommend withdrawal schedules....

This information is consistent with that in the LRA, except that the fifth capsule (Capsule W) of McGuire 1, which was in the RV before issuance of the license amendment for the LRA (i.e., NUREG-1772), was removed from the RV at 19.22 EFPY. The NRC staff confirmed that some withdrawn surveillance data had accumulated sufficient neutron fluence to cover plant operation to 54 EFPY. Regarding the withdrawal of Capsule W in 2004, Section 3.1.3.2.2, "Aging Management Programs," of NUREG-1772 states

> ... removal and testing of Capsule W will meet the withdrawal schedule criteria in [American Society for Testing and Materials] ASTM E185-82 ["Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"] for a 5-capsule withdrawal program and will provide additional relevant information for the behavior of the McGuire 1 RV during the period of extended operation. This is conservative and acceptable since the applicant is only required to remove four McGuire 1 surveillance capsules for testing to meet ASTM E185-82....

This NRC staff finding from NUREG-1772 indicates that the NRC staff had accepted the licensee's removal of Capsule W from the McGuire 1 RV at a neutron fluence of approximately 19.22 EFPY. After removal of Capsule W, the licensee conducted tests on weld specimens, but not on plate specimens. Untested specimens from Capsule W were stored for potential future testing or further irradiation (after reinsertion). This is acceptable because McGuire 1 is only required to remove four surveillance capsules for testing to meet the requirements of ASTM E185-82.

Conclusion

The NRC staff concludes that since the licensee had already withdrawn all required capsules in accordance with the requirements of ASTM E185-82 to support the 60-year license, there is no longer a need for Duke Energy to provide surveillance capsule withdrawal schedules for McGuire 1 and 2 in the LAR.

3.2.1.4 RV Internals and Core Support Structures

Regulatory Evaluation

The RV internals and core support structures include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCPB). The NRC staff's acceptance criteria for RV internals and core support structures are based on GDC 1, "Quality standards and records," and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RV internals and core supports. Matrix 1 of NRC Review Standard RS-001, Rev. 0, "Review Standard for Extended Power Uprates" (ADAMS Accession No. ML033640024), provides references to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (ADAMS Accession No. ML12335A511) and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (ADAMS Accession No. ML003708443).

Both reports for PWR RV internals were superseded by the Materials Reliability Program (MRP) Report 1022863 (MRP-227-A), "Pressurized Water Reactor Internals Inspection and Evaluation [I&E] Guidelines," dated December 2011 (ADAMS Accession No. ML12017A194), which also contains the NRC staff SE for this report. MRP-227-A provides the industry's recommended I&E guidelines for PWR RV internals as a result of the industry effort on this issue for the past few years.

Technical Evaluation

The licensee discussed the impact of the LAR on the structural integrity of the RV internals in Enclosure 2 of the LAR, Section IV.1.A.ii.

The NRC staff reviewed the licensee's evaluation of the structural integrity of the McGuire 1 and 2 RV internals using RS-001, Rev. 0. Matrix 1 of this document describes the NRC staff's basis for evaluating the potential for extended power uprates to induce aging effects on RV internals. Depending on the magnitude of the projected RV internals fluence, Matrix 1 may be applicable to evaluating the impacts of the LAR. However, WCAP-14577, Rev. 1-A, and BAW-2248-A, cited in Matrix 1, are no longer applicable after issuance of MRP-227-A, which summarized the most recent industry developed I&E guidelines for PWR RV internals, as modified by the associated NRC staff SE. Section IV.1.A.ii of Enclosure 2 of the LAR states that

> It is therefore concluded that there is no impact, adverse or otherwise, from the McGuire Units 1 and 2 MUR uprate on the plant-specific implementation of the MRP-227

> requirements. MRP-227, Rev. 0 (Reference IV-1 1) has not been approved by the NRC, and it will likely be revised to incorporate the NRC's comments. Should any future revisions of MRP-227 affect the MUR power uprate, they will be identified during review of the inspection plan and addressed by the appropriate process.

However, Section IV.1.A.ii did not mention meeting the requirements of MRP-227-A by specific dates during the current 40-year license to demonstrate that the degradation of the RV internals will be managed appropriately. This was the subject of the NRC staff's RAI question 42 (which superseded RAI question 18). RAI question 42 requested a specific confirmation from the licensee.

The licensee's response to RAI question 42, sent by letter August 15, 2012, confirmed the following:

(1) As required by Section 7.2 of MRP-227-A, the Aging Management Programs for McGuire Units 1 and 2 reactor internal components were developed and documented by December 31, 2011. These programs were documented in WCAP-17466-NP, Revision 0, December 2011 (Unit 1) and WCAP-17467-NP, Revision 0, December 2011 (Unit 2).

(2) McGuire commits to implement the guidelines of Section 7.3 of MRP 227-A no later than December 31, 2013, including the performance of inspections of applicable components within the time frame specified in Tables 4-3, 4-6, 4-9, and 5-3 of MRP-227-A. This commitment is documented in Enclosure 3 of this RAI response.

Based on the above, the NRC staff finds that the licensee's management of the RV internals is consistent with the industry's I&E guidelines documented in MRP-227-A, as modified by the associated NRC staff SE, and is, therefore, acceptable to the NRC staff. RAI question 42 is resolved, and the NRC staff accepts the licensee's evaluation of the effects of the LAR on the RV internals and core support structures.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the impact of the LAR on the structural integrity assessments for the RV internals. The NRC staff has determined that the licensee's RV internals evaluation considering the effect of the LAR is acceptable because (1) the licensee confirmed that the McGuire 1 and 2 RV internals aging management was developed before December 31, 2011, and the inspection program (plan) will be implemented by December 31, 2013, as required by the MRP-227-A and (2) the LAR would result in very small changes to aging parameters such as temperature and neutron flux.

3.2.1.5 Conclusion for RV Integrity and RV Internal and Core Support Structures

The NRC staff has reviewed the LAR and has evaluated its impact on the structural integrity assessments for the RV and RV internals. The NRC staff has determined that, with the license condition documented in its supplement to the LAR dated March 7, 2013, regarding P-T limits, the changes identified in the LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) PTS assessment; (2) P-T limits; (3) RV USE assessment; (4) RV surveillance program; and (5) RV internals and core support structures.

3.2.2 Instrumentation and Controls

3.2.2.1 Introduction

This MUR power uprate is based on the use of the feedwater flow measurement techniques of a Cameron (formerly Caldon) Leading Edge Flow Meter (LEFM) CheckPlus™ System. The LAR references the following topical reports: ER-80P and ER-157P, Rev. 8, and their respective SEs dated March 8, 1999, and August 16, 2010.

These two topical reports, which are generically applicable to nuclear power plants, document the ability of the Cameron LEFM CheckPlus Systems to increase the accuracy of flow measurement. ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Cameron LEFM Check System in a typical two-loop pressurized-water reactor (PWR) or two-FW-line boiling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. ER-157P, Rev. 8, and supplements describe the Cameron LEFM CheckPlus System and list the results of a typical PWR or BWR thermal power measurement uncertainty calculation using the Cameron LEFM CheckPlus System. Together, these two topical reports, along with the SEs approving them, provide the generic safety basis for an MUR power uprate.

The plant-specific bases for the proposed MUR uprate at McGuire 1 and 2 is described in more detail in proprietary appendices to Enclosure 2 to the LAR.

3.2.2.2 Regulatory Evaluation

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

As the maximum power level of a nuclear plant is a licensed limit, the NRC must review and approve a proposal to raise the licensed power level under the license amendment process. The LAR should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

ER-80P and ER-157, Rev. 8, describe the Cameron LEFM CheckPlus System for the measurement of FW flow and provide a generic basis for the proposed power uprate. The NRC staff also considered in its review the guidance contained in RIS 2002-03.

The guidance contained in RG 1.105, Rev. 3, describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the limits set by the plant's TSs. The method described in RG 1.105 for combining instrument uncertainties can be used for combining the uncertainties associated with the secondary calorimetric calculation. This allows licensees to justify a power uprate with reduced margin between the CLTP and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power.

3.2.2.3 Technical Evaluation

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

FW flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the FW velocity in the pipe. Because of the high cost of calibrating the venturi and the need to improve flow instrumentation measurement uncertainty, the industry evaluated other FW flow measurement techniques and found the Cameron LEFM CheckPlus UFMs to be a viable alternative.

3.2.2.3.1 LEFM Technology and Measurement

The NRC staff's review in the area of instrumentation and control covers the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique in accordance with guidelines in Section I of Attachment 1 to RIS 2002-03 (evaluated by the NRC staff in Section 3.1.1.2.1 of this SE above). The NRC staff conducted its review to confirm that the licensee's implementation of the proposed FW flow measurement device is consistent with the NRC staff-approved Topical Reports ER-80P and ER-157P, Rev. 8, and that the licensee adequately addressed the additional requirements listed in the NRC staff's SEs for these topical reports. The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.3 percent correctly accounts for all uncertainties associated with power level instrumentation errors, and (2) the uncertainty calculations meet the relevant requirements of 10 CFR Part 50, Appendix K.

The proposed Cameron LEFM CheckPlus System includes an electronic cabinet and four measurement spool pieces to be installed in each of the four main FW flow lines upstream of the existing FW venturi flow meters. Each measurement section consists of 16 ultrasonic, multipath, transit time transducers, and FW pressure input. The NRC staff has reviewed the licensee's LEFM locations proposed for McGuire 1 and 2 and finds that they meet the Cameron requirements for LEFM location and are acceptable.

The supplement to the LAR dated September 27, 2012, provided additional detail regarding how the LEFM will function at McGuire 1 and 2. Specifically, the licensee stated that the LEFM signal will be used to calibrate the signal from the ASME flow nozzles. When an LEFM system is non-functional, however, signals from an existing ASME flow nozzle will be used as input to the secondary calorimetric portion of the RTP calculation in place of the LEFM System.

In the LAR, the licensee stated that McGuire 1 and 2 will not consider a CheckPlus System with a single failure as a separate category; it will be considered as an inoperable LEFM and will be treated as an allowed outage (as described in more detail in Attachment 1 to RIS 2002-03, Section 1, Items G and H).

In a supplement to the LAR dated July 6, 2012, responding to NRC staff RAI question 20a, the licensee submitted ER-972, Rev. 2, which provides traceability between ER-157P-A and ER-80P and the plant-specific Cameron Engineering Reports ER-819, ER-822, ER-823, and ER-874. Further, the licensee provided detailed cross references between the plant-specific reports and the Cameron Topical Reports in Enclosure 3 to its supplement to the LAR dated July 6, 1012. The NRC staff reviewed these responses and verified them against the calculations in ER-874, ER-823, ER-822, and ER-819 and finds that the licensee has performed the plant-specific calculations in conformance with the methodology the NRC staff approved in approving ER-157P-A, Rev. 8 and Rev. 8 Errata.

In the LAR, the licensee stated that a safety analysis was performed for the McGuire 1 and 2, MUR uprate and that the analysis did not require any adjustment to the Reactor Trip System or ESF Actuation System nominal setpoints or allowable values from the non-uprated values.

3.2.2.4 NRC Staff Conclusions Regarding Instrumentation and Control

The NRC staff reviewed the LAR, as supplemented, with respect to its analysis of instrumentation and control issues. Based on its review of the licensee's LAR, and supplements to the LAR dated July 6, 2012, September 27, 2012, November 1, 2012 and March 7, 2013, including uncertainty calculations, and referenced topical reports, the NRC staff finds that the licensee's proposed amendment is consistent with the approved Cameron Topical Report ER-80P and its supplemental Topical Report ER-157P. The NRC staff also finds that the licensee adequately accounted for instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations (as evaluated in Section 3.1.1.2.1 of the SE above). Therefore, the licensee's proposed amendment meets the relevant requirements of 10 CFR 50, Appendix K. On the basis of these considerations, the NRC staff finds the instrumentation and control aspects of the proposed thermal power uprate of approximately 1.7 percent acceptable.

3.2.3 Mechanical and Civil Engineering

3.2.3.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power, referred to as the CLTP. The regulation at 10 CFR Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing ECCS analyses for LOCAs. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in these analyses. The regulation at 10 CFR Part 50, Appendix K, allows licensees to assume a power level less than 1.02 times the licensed power level (but not less than the licensed power level) "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." As previously stated, the licensee has proposed to use a power measurement uncertainty of 0.3 percent based on the installation of the Cameron CheckPlus LEFM system. This system provides a more accurate measurement of FW flow than current systems, including those available when 10 CFR Part 50, Appendix K, was issued.

The NRC staff's review of the LAR in the areas of mechanical and civil engineering focused on verifying that the licensee has provided reasonable assurance that the structural and pressure boundary integrity of SSCs at McGuire 1 and 2 will continue to be adequately maintained following the implementation of the LAR under normal and abnormal loading conditions. Reasonable assurance is provided by demonstrating compliance with the NRC regulations listed below, which address the mechanical and civil engineering scope of the NRC staff review.

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

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

The design and licensing bases for the facility establish the principal means by which the facility demonstrates compliance with applicable NRC regulations. As such, the NRC staff's review primarily focused on verifying that the design and licensing basis requirements related to the structural and pressure boundary integrity of SSCs affected by the LAR would continue to be satisfied at LAR conditions. This, in turn, provides reasonable assurance that continued compliance with the applicable regulations will be maintained. Section 3.1 of the UFSAR describes how the facility complies with the GDC.

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

The NRC staff has recently issued similar MUR power uprate license amendments for the Surry Power Station, Units 1 and 2, on September 24, 2010, (ADAMS Accession No. ML101750002), Prairie Island Nuclear Generating Plant, Units 1 and 2, on August 18, 2010, (ADAMS Accession No. ML102030573), and for the North Anna Power Station, Units 1 and 2, on October 22, 2009 (ADAMS Accession No. ML092250616).

3.2.3.2 Technical Evaluation

The NRC staff's review in the areas of civil and mechanical engineering focused on the impact of the LAR on the structural and pressure boundary integrity of the pressure-retaining components and associated component supports, including piping and pipe supports. The civil and mechanical engineering review scope also evaluated any impact of the LAR on the structural integrity of the RV internals, including core support structures and non-core support structures. In addition, the NRC staff's review considered the impact of the proposed MUR power uprate 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 LAR on moderate energy pipe rupture locations was also performed. The NRC staff's review verified that the licensee has provided reasonable assurance of the structural and pressure boundary integrity of the aforementioned SSCs and their supports under normal and abnormal loading conditions, including those due to postulated accidents and natural phenomena, such as earthquakes.

The proposed 1.7 percent power uprate will increase the RTP level from 3411 MWt to 3469 MWt at McGuire 1 and 2. In accordance with the requirements of 10 CFR Part 50, Appendix K, the licensee notes in Section IV of Enclosure 2 to the LAR that the current ECCS (AORs) are based on a core power level of 102 percent of the CLTP of 3411 MWt. As such, the licensee has previously performed these analyses assuming a power level of 3479 MWt and the implementation of the proposed MUR power uprate would revise the licensed thermal power to a level lower (i.e., 3469 MWt) than that for which the licensee has already analyzed.

Power Uprate Evaluation Parameters and Design Bases

Table IV-1 in Section IV.1 of Enclosure 2 to the LAR shows the pertinent temperatures, pressures, and flow rates for the current and uprated conditions. The licensee evaluated the effects of the LAR at a bounding power level of 3479 MWt. This power level corresponds to the proposed level following the LAR (i.e. 3469 MWt) plus the revised uncertainty of 0.3 percent. As shown in the table, there is no change in the RCS operating pressure (2250 pounds per square inch absolute (psia)) as a result of the LAR. The RCS mechanical design flow of 105,000 gallons per minute (gpm) also remains unchanged due to implementation of the LAR. At full power, the implementation of the LAR would yield a hot leg temperature of 614.6 °F, increasing from the current temperature of 614.1 °F, and a cold leg temperature of 555.6 °F, decreasing from the current temperature of 556.1 °F, which results in no change to the average RCS temperature. The main steam (MS) pressure decreases by 0.3 psia to 1020.7 psia at the LAR conditions and the MS steam flow increases from 15.1 million pounds per hour (Mlbm/hr) to 15.5 Mlbm/hr at the LAR conditions. The FW temperature will increase by 2 °F to 442 °F as a result of implementing the LAR.

The information related to the structural qualification of SSCs at McGuire 1 and 2 is contained in Chapter 3 of the UFSAR. The UFSAR describes the design criteria applicable to the McGuire 1 and 2 SSCs, including loads, load combinations and acceptance criteria stipulated by the applicable codes of record for these SSCs. Additional information regarding the design specifications, functional description, design loads and design code requirements for the reactor internals is located in Section 4.2.2 of the UFSAR. Throughout the LAR, the licensee notes that implementation of the LAR does not change current operating transients, nor does it introduce additional transients. As such, loads resulting from these transients that are used in the structural evaluations of SSCs are not affected. Similarly, the LAR has no effect on the deadweight and seismic loads of existing SSCs. The NRC staff finds, therefore, that the loads used in the existing AORs for these SSCs remain valid.

The functional description of the RCS, including the RV, RCPs, RCS piping and SGs is discussed in Chapter 5 of the UFSAR. Chapter 10 of the McGuire 1 and 2 UFSAR provides the design basis information for the secondary side systems, including the MS and the FW and condensate system. In its supplement to the LAR dated June 21, 2012, the licensee confirmed that for SSCs within the scope of its license renewal efforts, the structural evaluations and analyses performed to support the LAR were performed consistent with the methodologies outlined in NUREG-1772.

<u>Pressure-Retaining Components and Component Supports</u>

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

The evaluations discussed in Section IV of RIS 2002-03 focus on determining what impact the MUR power uprate would have on the AOR for a particular SSC in order to determine whether the AOR for a particular SSC needs to be revised as a result of the power uprate. If the AOR for a particular SSC was performed using conditions which bound those which will be present at the MUR power level, no further evaluation is required. The design codes of record for the McGuire 1 and 2 RCS are documented in Table IV.1.D-1 of Enclosure 2 to the LAR. The licensee confirmed that MUR evaluations did not include any changes to the tabulated design codes of record. While the codes of record for the balance-of-plant (BOP) piping systems were not included in the aforementioned table, the licensee confirmed in its supplement to the LAR dated June 21, 2012, that BOP piping evaluations used the design basis codes of record.

The pressure-retaining components and component supports, including piping and pipe supports, which must be evaluated in support of an MUR power uprate include the following: the reactor pressure vessel (RPV), including the RPV shell, RPV nozzles and supports; the pressure-retaining portions of the control rod drive mechanisms (CRDMs); NSSS piping, pipe supports and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, secondary side internal support structures and nozzles; the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles and the surge line; and safety-related valves. Furthermore, Section IV.1.B of RIS 2002-03 indicates that the evaluation of those SSCs whose AOR are affected by implementation of an MUR power uprate,

> ... 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 MUR power uprate LAR. Affected components and supports refer to those for which their AOR is not bounded at MUR power uprate conditions. Pressure-retaining components and their supports generally remain unaffected by the implementation of an MUR power uprate based on the fact that they have been analyzed at conditions which are more limiting than those which will be present at MUR power uprate conditions (i.e., bounded). The licensee was able to disposition a number of components and their associated supports as unaffected by the proposed MUR power uprate, based on whether the plant parameter changes resulting from implementation of the LAR, identified above, affect the loads included in the AOR for the component and its supports. Based on its evaluations of the impact of LAR implementation on the components identified above, the licensee stated that the existing AORs related to the structural and mechanical qualifications of the following SSCs are unaffected by the proposed MUR power uprate at McGuire 1 and 2: the RPV, RPV nozzles and RPV supports; the pressure-retaining portions of the CRDMs; RCS piping and supports and loop branch nozzles; pressurizer shell, nozzles and surge line; the replacement SGs, including the shells, nozzles and secondary side internal support structures; and the pressure-retaining portions of the RCPs.

During its review of the LAR, the NRC staff issued two RAI questions regarding the evaluations performed for BOP piping and safety-related valves. In its supplement to the LAR dated June 21, 2012, the licensee stated that it had considered the temperature, pressure and flow increases in BOP piping systems affected by the MUR power uprate. However, the licensee also indicated that any of these parameter increases in BOP piping systems remain bounded by the operating parameters considered in the current AOR for affected BOP piping systems. Similarly, the licensee confirmed that the MUR power uprate has no effect on the structural integrity of safety-related valves at McGuire 1 and 2 and that these also remain bounded by their current AOR. Based on these considerations, the NRC staff concludes that all pressure-retaining components and supports, including piping and pipe supports, remain bounded at MUR power uprate conditions.

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

RV Internals

In accordance with Section IV.1.A.ii of RIS 2002-03, the licensee evaluated the effects of the proposed MUR power uprate on the McGuire 1 and 2 RV internals (RVIs). As discussed above, Section IV.1.B of RIS 2002-03 indicates that for those SSCs, including RVIs, whose AORs are affected by implementation of an MUR power uprate, the licensee should address the following, as they relate to the impact of the uprate on the AOR: stresses, cumulative usage factors (i.e., fatigue), flow-induced vibration (FIV), and changes in temperature, pressure and flow rates resulting from the power uprate. The licensee summarized its evaluation of the effects of the proposed power uprate on the structural integrity of the RVIs 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 RVIs due to the conditions which would be present following the implementation of the proposed MUR power uprate. The mechanical evaluations of FIV performed by the licensee are summarized in Section IV.1.B.iii of Enclosure 2 to the LAR. These evaluations focused on the potential for an increase in the vibratory response of the RVIs resulting from changes in the flow field at the MUR power level. An increase in vibratory response can introduce increased alternating stress intensities and subsequently higher cyclic fatigue of the RVIs. In Section IV.1.B.iii the licensee stated that

> Per the values in Table IV-1, the volumetric mechanical design flow remains unchanged for the MUR power uprate. Hence the vortex shedding frequencies remain unchanged. Also the temperature changes due to the MUR power uprate are less than 0.1 percent which causes a negligible change in the frequencies of the internals. Thus the stresses imparted on the RPV internals due to flow induced vibrations remain unchanged as a result of the MUR power uprate conditions and the existing analyses of record remain bounding.

Based on this evaluation, the licensee confirmed that the FIV characteristics of the RVIs are bounded by the current AOR.

The licensee's structural evaluations focused on determining whether the MUR power uprate induced loads on the RVIs greater than those for which the RVIs have been previously analyzed. Using these loads, the licensee is able to determine whether the design code requirements applicable to the RVIs will remain satisfied at the MUR power level. The NRC staff noted that Table IV.1.D-1 of Enclosure 2 to the LAR did not include the design code of record for the RVIs. In its supplement to the LAR dated June 21, 2012, addressing questions regarding the code of record for the RVIs, the licensee indicated that the design basis code of record for the RVIs was the January 1971 draft of Subsection NG of the ASME *Boiler and Pressure Vessel Code*, "Core Support Structures." The licensee also confirmed in its supplement to the LAR that the design basis code of record was used in evaluating the RVIs for acceptability at the MUR power level.

The NRC staff also issued an RAI question to the licensee regarding the licensee's structural evaluations of the RVIs, seeking clarification on the analytical uncertainties relied upon in its evaluation of the RVIs at the MUR power level. In its supplement to the LAR dated June 21, 2012, the licensee stated that

> The AOR related to the structural evaluation of the RVIs, including all core support and non-core support structures, did not require a revision to support implementation of the LAR. The uncertainty relied upon in the AOR to demonstrate that the calculation tolerance available in the AOR sufficiently bounds the core parameters proposed in the LAR is based on the core power level. The Reactor Internals were originally designed to support a core power level of 3479 MWt with a licensed core power level of 3411 MWt, which allows for approximately 2 percent uncertainty in the core power level. The evaluations performed in support of the MUR uprate consider a power level of 3469 MWt with the design core power level remaining at 3479 MWt, providing for approximately 0.3 percent uncertainty in core power level.

Based on this assessment, the licensee noted that the RVIs remain bounded at MUR conditions and no revision to the AOR is required to support MUR implementation.

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

Postulated Pipe Ruptures and Associated Dynamic Effects

The licensee evaluated the effects of the proposed MUR power uprate on systems classified as high energy to determine whether any changes to the HELB AOR will result from the implementation of the power uprate. This assessment is summarized in Section IV.1.B.vii of Enclosure 2 to the LAR. As indicated in the summary of the licensee's assessment, the current

AOR for HELBs was reviewed to determine whether the MUR power uprate would have any impact on the current HELB AOR. The licensee concluded that because the temperature and pressure changes in high energy systems are considered nominal, no new HELB locations are required to be postulated as a result of MUR implementation. In its supplement to the LAR dated June 21, 2012, the licensee responded to an NRC staff RAI question regarding moderate energy line breaks by confirming that the MUR power uprate has no effect on moderate energy piping systems and, as such, no new moderate energy pipe cracks are required to be postulated.

The licensee summarized its assessment of the impact of MUR implementation on jet impingement and thrust forces (dynamic effects) in Section IV.1.B.viii of Enclosure 2 to the LAR. The NRC staff's review of this information resulted in the issuance of an RAI regarding the scope of the licensee's evaluation in this topic area. In its supplement to the LAR dated June 21, 2012, the licensee confirmed that it had evaluated the impact of MUR implementation on the dynamic effects resulting from currently postulated HELBs. The licensee confirmed that these are not affected by the implementation of the MUR power uprate due to the fact that the changes in the temperatures and pressures of these systems resulting from MUR implementation were within the bounds of the temperatures and pressures which have been previously evaluated.

The NRC staff has reviewed the licensee's evaluations related to determinations of pipe rupture locations and their corresponding dynamic effects and finds that the licensee's assessments performed for these areas are acceptable. This acceptance is based on the information discussed above, which demonstrates that the AORs related to HELBs, medium energy line breaks (MELBs), and dynamic effects resulting from postulated pipe ruptures will remain bounding under the proposed MUR power level. The NRC staff considers this conclusion reasonable given the small magnitude increases in temperature and pressure which accompany MUR implementation. Correspondingly, as discussed above, these small changes generally have no impact on pressure-retaining components such as piping. Additionally, the NRC staff finds the licensee's dynamic effects assessment is acceptable given that pressure and temperature generally govern dynamic effects consequences. Given that MUR implementation results in no significant pressure or temperature changes in systems with postulated breaks, the NRC staff considers the licensee's conclusion that there is no impact on current dynamic effects assessment reasonable and acceptable.

3.2.3.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on the structural and pressure boundary integrity of pressure-retaining components and supports and RVIs. Additionally, the NRC staff reviewed the licensee's assessment of the effects on the McGuire 1 and 2 HELB and MELB AORs, including associated dynamic effects. Based on the review above, the NRC staff finds the MUR power uprate acceptable with respect to the structural integrity of the aforementioned SSCs affected by the power uprate. This acceptance is based on the licensee's demonstration that the intent of the aforementioned regulatory requirements, related to the civil and mechanical engineering purview, will continue to be satisfied following implementation of the MUR.

Specifically, the licensee demonstrated that: 1) the structural and pressure boundary integrity pressure retaining components and supports, including piping and pipe supports, at McGuire 1 and 2 are not affected by the proposed MUR power uprate, as evidenced by the fact that their

AORs are unaffected; 2) the RVIs at McGuire 1 and 2 also remain unaffected, when considering the impact of MUR implementation on the FIV characteristics and structural integrity of the RVIs; and 3) the McGuire 1 and 2 AORs related to the postulation of HELB and MELB locations, including dynamic effects associated with these postulated pipe ruptures, remains unaffected by the proposed MUR power uprate. Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of SSCs at McGuire 1 and 2 will be adequately maintained following implementation of the MUR power uprate, such that the MUR power uprate will not preclude the ability of these SSCs to perform their intended functions.

3.2.4 Electrical Engineering

3.2.4.1 Regulatory Evaluation

The regulatory requirements which the NRC staff applied in this portion of its review include the following:

The regulation at 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants." This regulation requires that licensees establish programs to qualify electric equipment important to safety which are located in harsh environments.

The regulation at 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 an established period of time, and to recover there from.

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

3.2.4.2 Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. 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 LAR on the following electrical systems/components:

- AC Distribution System
- Power Block Equipment (Main Generator, Generator Circuit Breakers, Transformers, Isolated-phase bus duct)
- Direct Current (DC) system
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- SBO
- EQ Program

AC Distribution System

The AC Distribution System is the source of power for the non-safety-related and safety-related (essential) buses. According to the UFSAR, the AC sub-systems consist of the 13.8 kiloVolt (kV) and 6.9 kV normal auxiliary system, 4.16 kV essential auxiliary system, and 600 volt (V) normal and essential auxiliary systems. The licensee indicated in the LAR that the AC distribution system is bounded by the existing analysis and calculations of record for the plant.

In its supplement to the LAR dated July 16, 2012, which responded to NRC staff RAI question 32, the licensee provided details of the expected AC load increases due to the MUR power uprate and the HP turbine replacement project. The licensee stated that the load changes are deemed to be bounded by the existing analysis. The NRC staff finds that the percentage changes of the motor load increase due to the MUR power uprate are within the motor rated horsepower rating and, therefore, do not impact the loading margin of the existing equipment. In the same document the licensee responded to RAI question 33, regarding updated calculations of the AC distribution systems. The licensee provided a list of the AC electrical calculations of record that were reviewed to determine the impact of the LAR on the safety-related system. The review by the licensee determined that the AC distribution system has adequate capacity and will not be adversely impacted by the minor load increases.

Based its review of the LAR, as supplemented, the NRC staff finds that the minor AC load changes are at the 6.9 kV Normal Auxiliary Electrical Distribution System, and, therefore, will not adversely impact the loadings and voltages of safety-related buses. The NRC staff finds that the AC distribution system will continue to provide its safety function under LAR conditions.

Power Block Equipment

As a result of the LAR, the RTP would increase to 3469 MWt from the previous value of 3411 MWt. In its review, the NRC staff issued RAI question 1 regarding the maximum megawatts electric (MWe) generation expected at each unit and the portion of the 80 MWe of additional generating capacity associated with the MUR power uprate and the HP Turbine replacement, respectively. In its supplement to the LAR dated May 29, 2012, responding to RAI question 1, the licensee indicated that the maximum MWe generation for each McGuire unit would increase to approximately 1185 MWe after the MUR power uprate and HP turbine replacement – a 40 MWe per unit increase from the current 1145 MWe generating capacity. Of this increase, 20 MWe per unit would be associated with the MUR power uprate and 20 MWe per unit would be associated with the HP Turbine replacement. The maximum capacity of new generators is 1450 mega-voltamperes (MVA) at a power factor of 0.90 lagging. In its supplement to the LAR dated November 1, 2012, the licensee indicated that to meet the grid voltage requirements under the plant operating conditions after the power uprate, the mega-voltamperes-reactive (MVARs) requirement for each unit at the point of interconnection with the grid would be 469 MVAR lagging and minus (-) 297 MVAR leading, calculated based upon required power factor values of 0.93 lagging and 0.97 leading, respectively (per agreement with the Transmission System Operator). All grid studies and documented generator capabilities support the ability of the new generators to meet the system requirements. The NRC staff reviewed the licensee's evaluation and, considering the rating of new generators, it finds that each main generator has adequate rating to meet the grid MVARs requirements corresponding to the increased generation of 1185 MWe.

In its supplement dated May 29, 2012, responding to RAI question 2 regarding the verification of isolated-phase bus (IPB) capacity, the licensee confirmed that the 24 kV IPBs are rated at 40,000 ampere, forced cooled, and can carry the maximum capacity of the new generators. The NRC staff finds that the IPBs and the main transformers (two 750 MVA, total 1500 MVA for each generator), remain adequately sized for the LAR conditions.

In its supplement to the LAR dated July 16, 2012, responding to RAI question 34 regarding the impact on protective relaying due to the increase in the main generator rating, the licensee indicated that the main generator stator replacement project will have an impact on the settings of the generator protective relays since the current settings are based upon the current generator rating of 1330 MVA (McGuire 1) and 1380 MVA (McGuire 2) versus the increased rating of 1450 MVA for both units following generator stator replacement. The licensee further stated that the new settings for the generator protective relaying have been determined and incorporated into approved calculations. Based on the licensee's response, the NRC staff has no further concern with respect to RAI question 34.

The licensee also reviewed the calculation related to the Generator Circuit Breakers (each rated at 20,000 ampere according to the McGuire 1 and 2 UFSAR, Section 8.3.1.1.2.1) and determined that the rating of each Generator Circuit Breaker will remain bounding for the power uprate conditions. The NRC staff finds that each Generator Circuit Breaker remains adequately sized for the power uprated conditions.

Based on review of the LAR, as supplemented by letters dated May 29, 2012, July 16, 2012, and November 1, 2012, the NRC staff finds that the licensee has adequately addressed the impact of the MUR power uprate conditions on the Power Block Equipment, and that the Power Block Equipment will have adequate capacity.

<u>DC System</u>

According to the McGuire 1 and 2 UFSAR, the DC systems consist of the Switchyard 125 V DC System, 250 V DC Auxiliary Power System, 125 V DC and 240/120 V AC Auxiliary Control Power Systems, and safety-related 125 V DC and 120 V AC Vital Instrument and Control Power Systems. The licensee indicated in the LAR that the DC systems are bounded by the existing analysis and calculations of record for the plant.

In its supplement to the LAR dated July 16, 2012, responding to RAI questions 35 and 37 regarding DC load increases due to the MUR power uprate and HP Turbine replacement, the licensee indicated that the existing margin in DC systems can accommodate the additional electrical load (new Cameron instrumentation system), which is battery-backed, non-vital, and 120 VAC source fed. In its response to RAI question 38 regarding the evaluation of the adequacy of the DC system, the licensee provided a list of the DC system-related electrical calculations of record that were reviewed to determine the impact of the LAR on the safety-related DC systems. The review by the licensee determined that existing DC distribution systems will not be adversely impacted. Since the impact of additional electrical load in the DC systems will be minimal; the NRC staff finds that the existing DC systems will continue to perform their design function under MUR power uprate conditions.

EDGs

The 4.16 kV Essential Auxiliary Power System (EPC) provides emergency electrical power for the plant engineered safety features (ESFs) plus selected BOP emergency loads in the event that the normal AC power is interrupted. According to Section 8.3.1.1.7 of the UFSAR, the EPC consists of two full capacity EDGs rated at 4000 kW per unit. In the LAR, the licensee indicated that the LAR will not change the safety-related loads of the EDGs.

Since there is no increase in safety-related loads, the NRC staff finds that the existing EDGs will continue to perform their design function under LAR conditions.

Switchyard

According to the UFSAR Section 8.2.1.2, McGuire 1 and 2 has 230 kV and 525 kV switchyards. McGuire 1 is connected to the 230 KV switchyard and McGuire 2 is connected to the 525 kV switchyard.

In its supplement to the LAR dated July 16, 2012, responding to RAI question 39 regarding any impact on the switchyard components due to the increase in power output from the MUR power uprate, the licensee indicated that the switchyard components were originally designed to accommodate the main power step-up transformer rating (McGuire 1 -1520 MVA, 230 kV; McGuire 2 -1500 MVA, 525 kV), which exceed the replacement generator's design rating of 1450 MVA. Therefore, the 230 kV and 525 kV switchyard components can accommodate the increased 40 MWe power output from each McGuire unit.

The NRC staff reviewed the information provided by the licensee, and finds that the existing switchyard equipment ratings have adequate margins and the switchyard system is capable of supporting the LAR conditions.

Grid Stability

In the LAR, the licensee indicated that system impact studies for the McGuire 1 and 2 power uprate for a total additional 80 MWe were performed, and concluded the following:

1) Thermal Analysis Study: The thermal study results, following the inclusion of the increased generation, were evaluated by the licensee through a process of annual screening according to Duke Energy Power Transmission System Planning Guidelines, to identify impacts on the system. The study concluded that no transmission network upgrades were necessary.

2) Stability Study: The licensee's study concluded that McGuire 1 and 2 can reliably inject an additional 80 MW of net power into the Duke Energy Carolinas electric transmission system without any stability issues. The study used criteria based on a North American Electric Reliability Corporation Transmission Planning Reliability Standard.

3) Fault Study: This study was not needed since the new generator reactances were slightly higher than the existing reactances, thus lowering the fault current. Therefore, the additional power output will not have an impact on the existing fault study.

4) Reactive Capability Study: The study concluded that with the proposed modifications to the existing generating facility, adequate reactive support will continue to exist in the region.

In its supplement to the LAR dated July 16, 2012, in responding to NRC staff RAI question 40 regarding the degraded voltage relay settings, the licensee indicated that the minimum McGuire 1 and 2 switchyard voltages are based upon the degraded voltage relay settings at the safety-related buses (as specified in TSs Surveillance Requirement (SR) 3.3.5.2.) and plant loading. The degraded voltage relay settings at the safety-related buses will not be changed under LAR conditions. Therefore, any changes in switchyard voltages after LAR implementation will be dependent on plant loading. Any change in these LAR switchyard voltages will be discussed and communicated with the Transmission System Operator prior to implementation of the MUR. In its supplement to the LAR dated January 2, 2013, the licensee provided the following schedules for installation of Check Plus LEFM system, HP Turbine and Generator/Exciter Replacements, and MUR implementation for McGuire 1 and 2:

McGuire 1 – LEFM system installation and HP Turbine Replacement during the spring 2013 Refueling Outage (RFO); Generator/Exciter Replacement in the fall 2014 RFO. In its supplement to the LAR dated January 2, 2013, the licensee accepted the following license condition regarding McGuire 1:

> [McGuire Nuclear Station] MNS switchyard voltages required (so as not to impact the degraded voltage relay settings), corresponding to [McGuire Nuclear Station] Unit 1 post-MUR uprate conditions, will be evaluated prior to implementation of MUR on Unit 1. However, if at the time of this evaluation, Unit 1 is not capable of realizing the expected maximum post-MUR uprate MWt power level and/or Unit 1 is not capable of generating the expected maximum post-MUR uprate MWe, then an additional evaluation will be performed when Unit 1 has these capabilities. If this additional evaluation is necessary, any changes in the [McGuire Nuclear Station] MNS switchyard voltages required (so as not to impact the degraded voltage relay settings), corresponding to conditions associated with the additional Unit 1 MWt capability and/or the additional Unit 1 MWe capability, will be evaluated prior to raising Unit 1 reactor core full steady state power to the expected maximum post-MUR uprate MWt power level and/or prior to Unit 1 generating the expected maximum post-MUR uprate MWe.

McGuire 2 – HP Turbine and Generator/Exciter Replacements and LEFM system installation occurred in the fall 2012 RFO. The MUR implementation is currently scheduled to be implemented during the McGuire 2 fuel cycle which started in November 2012 and is scheduled to end in March 2014.

In its supplement to the LAR dated January 2, 2013, the licensee accepted the following regulatory commitment regarding McGuire 2:

> Any changes in the MNS switchyard voltages required (so as not to impact the degraded voltage relay settings), corresponding to the Unit 2 post-MUR uprate, Unit 2 HP turbine Replacement, and Unit 2 Generator Stator/Exciter Replacement conditions, will be evaluated.

The commitment date for this commitment is "prior to implementation of the MUR at McGuire 2."

Based on review of the LAR and the information in the supplement to the LAR dated July 16, 2012, and January 2, 2013, as well as the license condition developed for McGuire 1 and regulatory commitment provided for McGuire 2, the NRC staff finds that the licensee has adequately addressed the impact of the LAR conditions on the grid stability related studies. The offsite power system will continue to be acceptable corresponding to the LAR conditions.

Station Blackout

For McGuire 1 and 2, the SBO scenario assumes that both units experience a loss of offsite power and that one unit's EDGs completely fail to start. At least one EDG is assumed to start for the non-SBO unit. The SBO coping duration for McGuire 1 and 2 is four hours. This is based on the evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 as discussed in the McGuire 1 and 2 UFSAR, Section 8.4. In the LAR, the licensee stated that an alternate AC (AAC) source is provided at McGuire 1 and 2. The AAC source is the Standby Shutdown Facility (SSF) diesel generator (DG). The SSF DG is available within 10 minutes of an SBO event. The SSF DG has sufficient capacity and capability to operate equipment necessary to maintain a safe shutdown condition for the four-hour SBO event.

The licensee evaluated the impact of the LAR on the AAC source, condensate storage tank inventory, Class 1E battery capacity, compressed air, containment isolation, and RCS inventory currently credited for SBO mitigation. In the LAR, the licensee indicated that these mitigating items are not impacted by the LAR. In particular, McGuire 1 and 2 has four Class 1E batteries which are shared between the units. There are five battery chargers on site, one for each battery and one spare charger, each of which can be powered from either unit. Considering that the battery chargers can be fed from the non-SBO unit, McGuire 1 and 2 has sufficient battery capacity to cope with a 4-hour SBO unit.

The NRC staff finds that the licensee has adequately evaluated the impact of the LAR on SBO coping duration, and that the LAR will not impact the McGuire 1 and 2 SBO coping duration of four hours. The licensee will continue to meet the requirements of 10 CFR 50.63 under LAR conditions.

Environmental Qualification Program

In the LAR, the licensee described the McGuire 1 and 2 EQ Program as guided by the regulation at 10 CFR 50.49, as implemented by the guidance of NUREG-0588, Rev. 1 (ADAMS Package Accession No. ML112990731). In its supplement to the LAR dated July 6, 2012, which responded to NRC staff RAI question 4 regarding the completeness of the EQ evaluation, the licensee stated that:

> The review of McGuire Nuclear Station (MNS) EQ Program documentation included review of both Duke Energy EQ program-level documents and discrete EQ files/calculations for specific components installed at MNS. This review was conducted to focus on the EQ parameters of temperature, pressure, and radiation, with respect to any potential parameter changes due to the MUR power uprate.

Based on its evaluation of the temperature and pressure, the licensee determined that the BOP systems showed some slight parameter changes, but these minor changes would have no impact on the EQ components at McGuire 1 and 2. The evaluation of the systems inside Containment and in the Doghouse for accident temperature and pressure conditions showed that the current design basis analyses were performed at 102 percent RTP which bounds the MUR uprate. The licensee evaluations showed no EQ impact with respect to temperature or pressure due to the MUR uprate.

In its supplement to the LAR dated July 6, 2012, the licensee stated that

> The potential impact of the MUR uprate on radiation dose was evaluated for MNS EQ equipment in the equipment data base. No items were identified that were impacted by the MUR power uprate dose change (i.e. were qualified for the pre-MUR Total Integrated Dose but not qualified for the post-MUR Total Integrated Dose).
>
> This review evaluated one existing, operable but degraded/non-conforming condition (OBDN) and one operability issue. Certain reactor vessel level indication RTDs were previously determined to be OBDN dependent on their installed location. In addition, one of three MNS Unit 1 pressurizer level transmitters was determined to be inoperable for post accident monitoring but operable for other required normal operation functions. Resolution of these existing conditions, which are being tracked in the corrective action program, are applicable for current operating conditions and conditions after MUR implementation.

In its supplement to the LAR dated September 27, 2012, the licensee provided a regulatory commitment which stated

> Unit 2 reactor vessel level indication RTDs (2NCRD8360 and 2NCRD8430) and a Unit 1 pressurizer level transmitter (1 NCL T5170), previously identified in the July 6, 2012 MNS response to MNS MUR LAR RAI Question 4 as being in a degraded/non-conforming condition and an inoperable condition respectively, will be returned to an operable condition prior to implementation of the MUR power uprate on the applicable MNS Unit.

The commitment date for this commitment was "Prior to implementation of the MUR power uprate on the applicable MNS Unit."

The licensee determined that no specific component modifications are required to support the LAR. The NRC staff finds that the licensee has adequately evaluated the impact of the LAR on the EQ component and that the LAR will have no adverse impact on the licensee's ability to continue to meet the EQ requirements of 10 CFR 50.49 as implemented by the guidance of NUREG-0588, Rev. 1.

3.2.4.3 Conclusion For Electrical Engineering Section

The NRC staff reviewed the licensee's technical evaluations described above and, based on that information, the NRC staff finds that McGuire 1 and 2 will continue to meet the requirements of 10 CFR 50.49, 10 CFR 50.63, and GDC 17. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to electrical engineering evaluations.

3.2.5 Chemical Engineering and Steam Generator Integrity

3.2.5.1 Chemical and Volume Control System

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

Regulatory Evaluation

The NRC staff has reviewed the safety-related functional performance characteristics of CVCS components. The NRC's review criteria are based on GDC 14, "Reactor coolant pressure boundary" and GDC 29, "Protection against anticipated operational occurrences." GDC 14 states that "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." GDC 29 states that "The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences." Specific review criteria are contained in the SRP, Section 9.3.4, "Chemical and Volume Control System (PWR)."

Technical Evaluation

The licensee in Section IV.1.A.v of Enclosure 2 to the LAR reviewed accidents, transients and other UFSAR analyses to determine the impact of the MUR on the CVCS. The licensee reported that the hot-leg and cold-leg temperatures of the RCS at RTP were projected to increase by 0.5 °F to 614.6 °F and decrease by 0.5 °F to 555.6 °F degrees, respectively.

The RCS pressure and average temperature are projected to remain the same, 2250 psia and 585.1 °F, respectively. The licensee evaluated the effects of the LAR on the CVCS and determined that the CVCS will continue to satisfy the design basis requirements when considering the temperature, pressure and flow rate effects resulting from the power uprate.

The NRC staff finds that the licensee has demonstrated that the CVCS will continue to maintain RCS inventory and water chemistry. The NRC staff also finds that the CVCS will continue to meet system design requirements and that no new design transients will be created at LAR conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed LAR on the CVCS and concludes that the licensee has adequately addressed changes to the reactor coolant and its effects on the CVCS. The NRC staff further concludes that the licensee has demonstrated that the AOR for the CVCS will continue to be acceptable and meet the requirements of GDC 14 and GDC 29 following implementation of the LAR. Therefore, the NRC staff finds the LAR acceptable with respect to the CVCS.

3.2.5.2 Steam Generator Blowdown System

Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SG blowdown system (SGBS) provides a means for removing SG secondary-side impurities and, thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. 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's review criteria for the SGBS are based on GDC 14, as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture.

Technical Evaluation

The licensee evaluated the SGBS in Section IV.1.A.v of Enclosure 2 to the LAR and determined that it will continue to function within its design basis at LAR conditions. In its supplement to the LAR dated July 6, 2012, the licensee stated that the SGBS

> ... operates continuously with the system flow rate set based on plant chemistry requirements. Any increase in Blowdown System flow rate caused by potentially higher impurity content under MUR conditions would be bounded by the increase in overall secondary side flow of 2.4 percent resulting from the MUR. Therefore, the Blowdown System was evaluated conservatively with a bounding increase in flow of 2.4 percent. The evaluated 2.4 percent increase in Blowdown flow at the uprate conditions remain below the current design flow of the system. The Steam Generator Blowdown System will continue to perform its intended function given the potentially higher flow and impurity content under the proposed MUR conditions.

Furthermore, the licensee stated that

> The components of the Stream Generator Blowdown System susceptible to flow-accelerated corrosion [FAC] will continue to be managed in accordance with the FAC Program. The MUR Power Uprate will not result in the removal of components currently managed in the FAC Program.

The NRC staff reviewed the licensee's UFSAR and confirmed that LAR conditions will continue to be bounded by the current licensing basis for the SGBS.

Conclusion

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

3.2.5.3 Steam Generator Tubes

Regulatory Evaluation

SG tubes constitute a large part of the RCPB. As a result, their integrity is important to the safe operation of a reactor. The NRC staff's review in this area covered the effects of changes in operating conditions resulting from the LAR on SG materials and the SG program. The NRC staff's review criteria for the SG Program are based on the McGuire 1 and 2 TSs. Specific review criteria for this topic are contained in the SRP, Section 5.4.2.1, "Steam Generator Materials," (ADAMS Accession No. ML070380192) for the SG materials, and Section 5.4.2.2, "Steam Generator Program," (ADAMS Accession No. ML070380194) for the SG program.

The review guidance in the SRP, Section 5.4.2.1 is provided

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

The review guidance in the SRP, Section 5.4.2.2 is provided

> ... 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.

Technical Evaluation

McGuire 1 and 2 each have 4 Babcock & Wilcox International, Inc. (BWI) Model CFR-80 SGs. Each SG contains 6,633 thermally-treated Alloy 690 tubes. The licensee's evaluation of the SG tubes is described in Section IV.1.A.vi of Enclosure 2 to the LAR.

In this section the licensee stated that

> The MUR conditions were reviewed for impact on the existing design basis analyses for the steam generators. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are

within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the steam generator remain applicable for the uprated power conditions.

The NRC staff evaluated the material provided by the licensee. The NRC staff finds that the changes in operating conditions at LAR conditions would be small. In fact, the new operating temperatures and pressures are typical of those used by other plants with recirculating SGs, 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 concludes that the materials used in the SG remain acceptable, the fracture toughness of the ferritic materials is adequate, the design still limits the susceptibility of the materials to degradation and corrosion, the materials used in the SG remain compatible with the environment, the design permits the removal of impurities, and that any degradation that could occur is either avoided or can be managed.

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

With respect to the tube repair criteria included in the TSs for the SG program, the small changes in operating conditions are expected by the NRC staff to have a small, if any, effect on the structural limits for the tubes. Since the tube repair criterion is determined from the structural limit, it may also be slightly affected by the LAR conditions. Although this analysis was not reviewed by the NRC staff in detail, the NRC staff concludes that the tube repair criteria remain valid under the LAR conditions. This conclusion is based on NRC staff approval of identical 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 concludes that the SG program remains acceptable for the LAR conditions.

Conclusion

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

3.2.5.4 Flow-Accelerated Corrosion (FAC)

FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel are very resistant to FAC, and FAC is significantly reduced in components containing even small amounts 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.

Regulatory Evaluation

The NRC staff reviewed the effects of the LAR on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of damaged components could be made before reaching a critical thickness. The NRC staff's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

The licensee's evaluation of FAC is found in Section IV.1.E of Enclosure 2 to the LAR. The licensee's evaluation shows that the MUR power uprate will not have a significant impact on the licensee's FAC Program. The licensee anticipates that the FW system will experience the largest increase in wear. However, even this increase may be undetectable. The licensee will determine the impact of the MUR on the future piping wear rates through the use of modeling software.

During its review of the licensee's evaluation of FAC, the NRC staff issued RAI question 23. In its supplement to the LAR dated July 6, 2012, responding to questions about FAC, the licensee stated that "The purpose of the [FAC] program is to monitor piping systems that are subject to FAC degradation, and to mitigate pipe wall loss. The FAC program is based on the guidance of [Electric Power Research Institute] EPRI NSAC-202L-R3."

In its supplement to the LAR dated July 6, 2012, the licensee indicated that it had performed a preliminary wear rate analysis and provided these results in Enclosure 4 to the licensee's supplement to the LAR dated September 27, 2012. The licensee stated in this supplement that

> A wear rate analysis has been performed to assess the impact of the MUR on susceptible FAC components and sample results are shown in the Tables in Enclosure 4. They provide a comparison of the pre-MUR and post-MUR wear rates. Per this analysis, the increase in wear rates due to the MUR power uprate is considered minor and the existing FAC Program is adequate to incorporate the updated predictions.

The information provided in Enclosure 4 shows comparisons of pre-MUR and predicted post-MUR wear rates for those systems expected to experience the greatest increase in wear. Based on the analysis, the system that is anticipated to experience the greatest increase in wear rate is the heater bleed system followed by the FW system. The wear rate for the heater bleed system due to the MUR is projected to increase by 5.7 percent and 7.8 percent in Units 1 and 2, respectively. However, the licensee reported that the remaining FAC susceptible portions of the heater bleed system in both Units are scheduled for piping replacement by end of McGuire 1 and 2 operating cycle 23. In addition, the licensee stated that its model for monitoring these changes will be updated in order to incorporate the system changes associated with the MUR.

The NRC staff has reviewed the licensee's evaluation and supplemental information and finds that the current FAC program provides adequate margin to ensure that components susceptible to FAC are managed appropriately prior to exceeding minimum wall thickness. The NRC staff finds the increase in wear rate due to the LAR to be minimal. The NRC staff finds that the updated FAC program, with the incorporated system changes resulting from the LAR, will provide reasonable assurance that components susceptible to FAC will be managed appropriately after LAR implementation.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the impact of the LAR on the FAC analysis and concludes that the licensee has adequately addressed the impact of changes in plant operating conditions on the FAC analysis. The NRC staff finds that the licensee's modeling of FAC, combined with its ultrasonic inspections of components, provide adequate margin between actual component wall thicknesses and their minimum required design thickness. The NRC staff has reasonable assurance that the program will continue to be an acceptable predictive model after the implementation of the LAR. Additionally, the NRC staff concludes that the licensee has demonstrated that the updated analyses will predict, with reasonable assurance, the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the LAR. Therefore, the NRC staff finds the LAR acceptable with respect to FAC.

3.2.5.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination by radionuclides. Coatings also provide wear protection during plant operation and maintenance activities.

The NRC staff's review addressed the use of protective coating systems inside containment (Service Level I coatings) for their suitability and stability under design-basis LOCA conditions, considering radiation and chemical effects. The NRC staff's review criteria for protective coating systems are based quality assurance requirements of 10 CFR Part 50, Appendix B. The NRC staff also used RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Rev. 1 (ADAMS Accession No. ML003714475), for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in the SRP, Section 6.1.2.

Technical Evaluation

The licensee evaluated the impact of the MUR on its containment coatings program in Section VII.6.B of Enclosure 2 to the LAR. When reviewing this section, the NRC staff issued RAI question 27, regarding this program. In its supplement to the LAR dated July 6, 2012, the licensee in response to this RAI question stated that:

> Containment design pressure and temperature profiles were used to qualify the Service Level 1 coatings. The proposed power uprate does not change the current Design Basis

> Accident temperature and pressure profiles. Therefore, the coating qualification temperature and pressure profiles used to qualify the original maintenance Service Level 1 coatings continue to bound the Design Basis Accident temperature and pressure profiles under the proposed power uprate conditions.

The NRC staff has reviewed the licensee's evaluation and the UFSAR and has confirmed that the applicable regulatory guidance was followed. The NRC staff has reasonable assurance that the coatings will not be adversely impacted by the LAR and that temperature, pressure, and radiation limits under LAR conditions continue to be bounded by the conditions to which the coatings were qualified.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the LAR on protective coating systems. The NRC staff concludes that the licensee has appropriately addressed the impact of changes in conditions following a design basis LOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the LAR. Specifically, the protective coatings will continue to meet the requirements of 10 CFR Part 50, Appendix B, and the guidance in RG 1.54. Therefore, the NRC staff finds the LAR acceptable with respect to protective coatings systems.

3.2.6 Effect of Power Uprate on Major Components

Safety-Related Valves

The NRC staff's regulatory evaluation review criteria for the safety-related valve analysis are based on 10 CFR 50.55a. The NRC staff also examined the overall design change and included plant-specific evaluations using Generic Letter(s) (GL) 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance" (ADAMS Legacy Accession No. 8906290082); GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (ADAMS Legacy Accession No. 9508110268); GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (ADAMS Legacy Accession No. 9609100488); and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" (ADAMS Legacy Accession No. 9609250096).

The licensee evaluated the impact of the MUR on the existing safety-related valves design basis analysis. No changes in RCS flow, design, or operating pressure would be made as part of the LAR. The licensee's evaluations concluded that the temperature changes due to the LAR are bounded by those used in the existing analyses. As a result, none of the safety-related valves required a change to their design or operation as a result of the LAR. The analyses also confirmed that the existing MS safety valves capacity is adequate for overpressure protection at LAR conditions and that the existing lift setpoints are unchanged. Due to the insignificant changes in temperature and operating pressure, none of the safety-related valves required a change to their design or operation as a result of the LAR. 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 LAR.

The licensee also evaluated the impact of the LAR on the current air-operated valve (AOV) program, GL 89-10 & GL 96-05 motor-operated valve (MOV) program, and GL 95-07 pressure locking/thermal binding program. The overall system evaluations concluded that valve function, valve design, operational conditions, thrust, and torque requirements are unaffected by the LAR and all valves remain capable of performing their design basis functions. Therefore, no changes are required to the existing AOV, MOV, and pressure locking/thermal binding programs.

As part of its review, the NRC staff asked RAI question 19 regarding the licensee's AOV program. In its supplement to the LAR dated June 21, 2012, the licensee responded that the AOV Program for McGuire 1 and 2 is not impacted by the MUR and that the MUR does not alter the basis, scope, or content of the AOV Program.

Based on the licensee's evaluations, the NRC staff concluded that the performance of existing safety-related valves is acceptable with respect to the LAR and meets the regulatory requirements of 10 CFR 50.55a.

Safety-Related Pumps

The NRC staff reviewed the licensee's safety-related pump analysis. The NRC staff's criteria for reviewing the safety-related pumps analysis is based on the requirements in 10 CFR 50.55a.

The licensee evaluated the impact of the LAR on safety-related pumps in Section IV.1.A.v of Enclosure 2 to the LAR. The NRC staff reviewed the impact of the LAR conditions on the existing design basis analyses for safety-related pumps. The evaluation showed that there are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect pump performance. The current plant design is considered adequate and would require no modifications to pump systems.

On the basis of this information, the NRC staff concludes that the performance of existing safety-related pumps is acceptable with respect to the LAR and meets the regulatory requirements of 10 CFR 50.55a.

In-service Testing Program

The NRC staff reviewed the licensee's in-service testing (IST) program. The NRC staff's criteria for reviewing the licensee's IST program are based on the requirements in 10 CFR 50.55a.

In Section IV.1.E of Enclosure 2 to the LAR, the licensee described its evaluation of the impact of the MUR on the IST program for safety-related pumps and valves at McGuire 1 and 2. The Code of Record for McGuire 1 and 2 is the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1998 Edition through the 2000 Addenda. The IST program at McGuire 1 and 2 assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Therefore, the existing IST program will not be impacted by implementing the LAR.

The NRC staff reviewed the licensee's evaluation of its IST program and concludes that the IST program will be acceptable for the LAR conditions.

3.3 Safety Programs

3.3.1 Radiological Dose Assessment

Regulatory Evaluation

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

Past license amendments at McGuire 1 and 2 approved the implementation of the alternative source term for the fuel handling accident and LOCA in accordance with 10 CFR 50.67, "Accident source term." These past license amendments follow the guidance and methodology provided in applicable sections of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792) and SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (ADAMS Accession No. ML003734190).

For all other DBAs, the NRC staff evaluated the LAR against the requirements of 10 CFR 100, "Reactor Site Criteria," GDC 19, "Control Room," and applicable sections of the SRP.

Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of the LAR as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Sections II and III of Enclosure 2 to the LAR. The NRC staff also reviewed Chapter 15 of the UFSAR. The specific DBA analyses reviewed included

- MSLB (UFSAR Section 15.1.5)
- Locked Rotor Accident (UFSAR Section 15.3.3)
- Rod Cluster Control Assembly Misoperation (UFSAR Section 15.4.3)
- Rod Cluster Control Assembly Ejection Accidents (UFSAR Section 15.4.8)
- Instrument Line Break (UFSAR Section 15.6.2)
- SG Tube Rupture (UFSAR Section 15.6.3)
- LOCA (UFSAR Section 15.6.5)
- Radioactive Gas Waste System Leak or Failure (UFSAR Section 15.7.1)
- Radioactive Liquid Waste System Leak or Failure (UFSAR Section 15.7.2)
- Radioactive Releases Due to Liquid Tank Failure (UFSAR Section 15.7.3)
- Fuel Handling Accident (UFSAR Section 15.7.4)

The licensee indicated that the MSLB source term is independent of power level and that the thermal hydraulic analyses were performed initially at 0 percent power and are, therefore, unaffected by the MUR power uprate. In the application, the licensee also indicated that the current DBA dose AORs for the Locked Rotor Accident, Rod Cluster Control Assembly

Misoperation, Rod Cluster Control Assembly Ejection Accidents, LOCA, and Fuel Handling Accident were performed at 102 percent of the CLTP of 3411 MWt. The licensee further indicated that the McGuire 1 and 2 Instrument Line Break, SG Tube Rupture, Radioactive Gas Waste System Leak or Failure, Radioactive Liquid Waste System Leak or Failure, and Radioactive Releases Due to Liquid Tank Failure analyses results are independent of the assumed reactor power levels.

If the LEFM system described in the LAR experiences operational loss of signal, the licensee has accounted for a potential increase in measurement uncertainty beyond that assumed in the radiological dose consequence analyses.

Upon the loss of signal from the LEFM, a correction factor in the high direction is applied to the main FW flow signal for up to 3 days. After 3 days, if the LEFM signal is not restored, then the unit power will be reduced to or below the previously licensed RTP to ensure that the current licensing basis dose consequence analyses remain bounding. The acceptability of the LAR, as it pertains to the assumption that the correction factor is conservative to account for any drift during the 3 days, is addressed in Sections 3.1.1 and 3.2.2 of this SE.

Using the licensing basis documentation and the information in the LAR, as supplemented by the licensee's letter of July 6, 2012, the NRC staff verified that the UFSAR Chapter 15 radiological analyses assumptions, along with previously approved LARs, bound the conditions for the proposed power uprate.

Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA dose consequence analyses at the proposed uprated power level. The NRC staff finds that operating McGuire 1 and 2 at the proposed uprated power level will continue to meet the applicable dose limits following implementation of the LAR. The NRC staff further finds reasonable assurance that McGuire 1 and 2, following implementation of the LAR, 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 LAR is acceptable with respect to the radiological dose consequences of the DBAs.

3.3.2 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions, nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat due to the LAR on the plant's safe shutdown analysis to ensure that the SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's review criteria for the fire protection program are

based on (1) 10 CFR 50.48, "Fire protection," (2) GDC 3, "Fire protection," and GDC 5, "Sharing of structures, systems, and components."
The regulation at 10 CFR 50.48 requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant.

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

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

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

Technical Evaluation

Duke Energy developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR uprated power level of 3469 MWt against the previously analyzed core power level of 3411 MWt.

The NRC staff reviewed Enclosure 2 to the LAR, including Section VII.6.A, "Fire Protection Program," and Section II.1.D, item 47, "Safe Shutdown Fire." The NRC staff also reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection" (i.e., approved fire protection program). The review covered the impact of the LAR on the results of the safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the LAR on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

The NRC staff's review of the Enclosure 2 to the LAR, Section VII.6.A and Section II.1.D, item 47, identified areas in which additional information was necessary to in order for the NRC staff to complete its review of the LAR. In its supplement to the LAR dated June 21, 2012, the licensee responded to a series of RAI questions regarding its fire protection program.

In its supplement to the LAR dated June 21, 2012, the licensee responded to RAI question 5 which requested the licensee to verify that the MUR power uprate will not require any new operator actions. The licensee responded that no new operator manual actions are required as a result of MUR power uprate.

The NRC staff reviewed the licensee's response and found it satisfactorily and, thus, RAI question 5 is considered resolved based on the response. The NRC staff notes that this SE does not approve any new or existing operator manual actions concerning the McGuire 1 and 2 fire safe shutdown analysis.

In its supplement to the LAR dated June 21, 2012, the licensee responded to RAI question 6 which requested the licensee to verify that the plant can meet the 72-hour requirements in both 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at LAR power uprate conditions. The license responded, in part, that

> A review of the impact of the MUR power uprate on the design of the safe shutdown system identified the following three potential impacts due to the small (< 2 percent) increase in decay heat:
>
> - As part of the safe shutdown system function, the Main Steam safety valves are credited to release steam from the Steam Generators to maintain hot shutdown conditions. The small increase in decay heat at MUR power uprate conditions would result in a slightly higher frequency of Main Steam safety valve cycling.
> - Upon commencement of the unit cooldown, the operator throttles the credited Main Steam Power Operated Relief Valves. The small increase in decay heat at MUR power uprate conditions would result in an incremental increase in valve opening position.
> - Auxiliary feedwater inventory is initially from a condensate grade source of water with approximately 16 to18 hours capacity. However, the regulatory required auxiliary feedwater source is from the plant main condenser intake and discharge embedded piping for McGuire Unit 2 and Unit 1, respectively. Each unit's respective embedded piping has a nominal 3 to 3-½ days supply of cooling water which is readily replenished by gravity flow from Lake Norman. This is a sufficient supply of cooling water to accommodate the small increase in decay heat at MUR power uprate conditions and ensure that McGuire continues to meet the 72-hour requirements contained in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L.

The NRC staff reviewed the licensee's response and concluded that it satisfactorily addressed RAI question 6. RAI question 6 is considered resolved based on the following. For the MUR conditions, the licensee reviewed its systems to obtain and maintain plant in cold shutdown condition. The licensee indicated that McGuire 1 and 2 can be placed in cold shutdown following a fire within the required 72 hours. Further, the licensee indicated that additional decay heat removal would not impact the ability to reach and maintain cold shutdown within 72 hours.

In RAI question 7, the NRC staff stated that

> Some plants credit aspects of their fire protection system for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). If McGuire 1 and 2 credit its fire protection system in this way, the LAR should identify the specific situations and discuss to what extent, if any, the LAR affects these "non-fire-protection" aspects of the plant fire

> protection system. If McGuire 1 and 2 do not take such credit, then the NRC staff requests that the licensee verify this as well. In your response discuss how any non-fire suppression use of fire protection water will impact the ability to meet the fire protection system design demands.

In its supplement to the LAR dated June 21, 2012, the licensee responded to RAI question 7 and responded in part that

> Current procedures use the McGuire Fire Suppression System for two non-fire-protection beyond design basis events. For a loss of feedwater/auxiliary feedwater event, the system may supply up to 300 gallons per minute to Steam Generators as a secondary side heat sink. The loss of feedwater/auxiliary feedwater event assumes an initial power level of 102 percent rated thermal power (RTP) so the MUR power uprate will not impact this non-fire-protection aspect of the McGuire Fire Suppression System. The Fire Suppression System may also be used to supply backup make-up water to the Spent Fuel Pool for a loss of Spent Fuel Pool level event. As discussed in UFSAR Section 9.1.3.1.1, the spent fuel pool heat load is limited by procedure prior to core offload and therefore the MUR power uprate will not impact this non-fire-protection aspect of the McGuire Fire Suppression System.

The NRC staff reviewed the licensee's response and concluded that it satisfactorily addressed RAI question 7 based on the following. The licensee identified the following two provisions to use other features of the fire protection system for non-fire protection functions beyond design-basis events. In the first beyond design-basis event, the fire water is utilized to backup the water supply for the SGs in the event of a loss of FW/auxiliary FW. In the second event, the fire protection system provides backup water to add inventory to the SFP. The licensee analyzed and concluded that all these beyond design-basis events crediting the fire protection system are unaffected by the LAR. Therefore, the NRC staff finds the response to RAI question 7 acceptable because the licensee's analysis concluded that all of the above functions of non-fire suppression uses of fire protection water are beyond design-basis and would not be affected by the LAR.

Based on the licensee's fire-related safe-shutdown assessment and responses to RAI questions, the NRC staff concludes that the licensee has adequately accounted for the effects of the LAR on the ability of the required fire protection systems to achieve and maintain safe-shutdown conditions. The NRC staff finds this aspect of the capability of the associated SSCs to perform their design-basis functions after implementation of the MUR acceptable with respect to fire protection.

<u>Conclusion</u>

Based on its review, The NRC staff has concluded that the LAR will not have a significant impact on the fire protection program or post-fire safe shutdown capability and, therefore, the NRC staff finds the LAR acceptable with respect to these analyses.

3.3.3 Human Factors

3.3.3.1 Regulatory Evaluation

Guidance for reviewing the licensee's human factors evaluation is available in RIS 2002-03. Attachment 1, Section VII, items 1 through 4, of RIS 2002-03 contains a standard set of questions for human factors reviews of MURs. 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 LAR. The scope of the review included the identified changes to operator actions, human-system interfaces, and procedures and training needed for the LAR.

3.3.3.2 Technical Evaluation

The NRC staff has developed a standard set of questions for review of human factors issues associated with the review of MUR LARs in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4. The licensee's evaluation of human factors issues that are sensitive to the power uprate is described in Sections VII.1 through VII.4 of Enclosure 2 to the LAR. The following sections evaluate the licensee's analysis of human factors issues in the LAR as clarified by its supplement to the LAR dated June 21, 2012.

Operator Actions

In Section VII.1 of Enclosure 2 to the LAR Duke Energy stated that "An evaluation was performed of the Operator Actions [associated with the LAR] and no impacts were identified." The licensee also evaluated time critical operator actions individually in system evaluations and against the McGuire 1 and 2 licensing analyses presented in Section II of Enclosure 2 to the LAR to ensure they remain bounded. The licensee determined that all time critical operator actions "remain unchanged following the MUR power uprate." Additionally, the McGuire 1 and 2 design change process ensures that any impacted normal, abnormal and emergency operating procedures not yet identified as having affected operator actions will be revised prior to the implementation LAR if required.

The NRC staff has reviewed the licensee's statements in the LAR and its responses to the RAI questions related to any impacts of the LAR on existing or new operator actions. The NRC staff finds that the statements provided by Duke Energy are in conformance with the guidance in Section VII of Attachment 1 to RIS 2002-03. The NRC staff concludes that the LAR will not adversely impact operator actions or their response times.

Emergency and Abnormal Operating Procedures

With respect to emergency operating procedures (EOPs) and abnormal operating procedures (AOPs), Duke Energy stated in Section VII.2.A of Enclosure 2 to the LAR that

> The proposed MUR power uprate will be implemented under the administrative controls of the McGuire Nuclear Station design change process. The design change process

ensures any impacted emergency and abnormal operating procedures are revised prior to the implementation of the power uprate.

In addition, as noted above, the licensee determined in Section VII.1 of Enclosure 2 to the LAR that all time critical operator actions "remain unchanged following the MUR power uprate." As a result, Duke Energy concluded that no changes to operator actions in either the EOPs or AOPs were needed and that the available times to perform operator actions would remain unchanged. The revisions to the EOPs and AOPs will be done to reflect the higher power level and minor setpoint changes, and will be made prior to MUR implementation.

The NRC staff has reviewed Duke Energy's evaluation of the effects of the LAR on the McGuire 1 and 2 EOPs and AOPs and concludes that the LAR does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon: (1) Duke Energy revisions to the EOPs and AOPs that will reflect the new power level and revised setpoints and (2) the minor changes to the EOPs and AOPs that will be reflected in the operator training program prior to LAR implementation. The NRC staff finds that the statements and commitments provided by Duke Energy are in conformance with Section VII of Attachment 1 to RIS 2002-03.

Changes to Control Room Controls, Displays, and Alarms

In Section VII.2.B of Enclosure 2 to the LAR, Duke Energy described a process whereby changes to control room controls, displays (including the Safety Parameter Display System), and alarms related to the proposed MUR power uprate would be identified and completed prior to MUR implementation. In its supplement to the LAR dated June 21, 2012, the licensee clarified that this process would be in accordance with Duke Energy's Engineering Directives Manual 601. Duke's review of plant systems indicated that only minor modifications that redefine the new 100 percent power are necessary. Duke also provided supplemental information related to the engineering change program in its supplement to the LAR dated June 21, 2012. The only proposed changes to the control room displays, alarms, and controls would be those that reflect the increased power level, but these will not impact the operator's ability to perform operator actions or the available times needed to complete certain operator actions during accident scenarios. All changes to the control room, including modifications involving the Cameron/Caldon LEFM system, will be reflected in the operator training program prior to MUR implementation.

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

Control Room Plant Reference Simulator and Operator Training

In Section VII.2.C and VII.2.D of Enclosure 2 to the LAR, Duke Energy described the modifications to the plant simulator associated with the LAR, as well as the changes made to the EOPs and AOPs to be included in the operator training program and integrated with the control room modifications. The licensee will make these changes prior to MUR implementation.

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

Conclusion

The NRC staff has completed its human factors review of Duke Energy's LAR and concludes that the licensee has adequately considered, or will consider, the impact of the LAR on operator actions, EOPs and AOPs, control room components, the plant simulator and operator training programs. The NRC staff also finds that the statements provided by Duke Energy are in conformance with Section VII of Attachment 1 to RIS 2002 03.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (77 FR 28630). 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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Date: May 16, 2013

LIST OF ACRONYMS

AAC	alternate AC
AAO	anticipated operational occurrences
AC	alternating current
AFW	auxiliary feedwater
AOP	abnormal operating procedure
AOR	analysis of record
AOT	allowed outage time
AOV	air-operated valve
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BOP	balance-of-plant
BWI	Babcock & Wilcox International, Inc.
CLRTP	Containment Leakage Rate Testing Program
CLTP	current licensed thermal power
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
DBA	design-basis accident
DC	direct current
DG	diesel generator
DBA	design-basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full power years
EOLE	end-of-life extension
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EPC	Essential Auxiliary Power System
EQ	environmental qualification
ESF	engineered safety feature
F	degrees Fahrenheit
FAC	flow accelerated corrosion
FIV	flow induced vibration
FW	feedwater
gpm	gallons per minute
HELB	high energy line break
HP	high-pressure
HVAC	heating, ventilation and air conditioning
I&E	inspection and evaluation
IPB	isolated-phase bus
kV	kiloVolt
LAR	license amendment request
LEFM	leading edge flow meter
LOCA	loss-of-coolant accident
LRA	license renewal application

LTOPS	low temperature overpressure protection system
M&E	mass and energy
Mlbm/hr	million pounds per hour
MOV	motor-operated valve
MRP	Materials Reliability Program
MS	main steam supply system
MSLB	main steam line break
MUR	measurement uncertainty recapture
MVA	mega-voltamperes
MVAR	mega-voltampere-reactive
MW	megawatt
MWe	megawatts-electric
MWt	megawatts-thermal
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NS	Nuclear Service Water System
NSSS	Nuclear Steam Supply System
OBDN	operable but degraded/non-conforming condition
OM	ASME Code for Operation and Maintenance of Nuclear Power Plants
PCT	peak clad temperature
Psia	pounds per square inch, absolute
Psig	pounds per square inch, gauge
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RTD	resistance temperature detector
RTP	rated thermal power
RT_{NDT}	reference temperature for non-ductile transition
RT_{PTS}	reference temperature for PTS
RV	reactor vessel
SB	Main Steam Vent to Condenser
SBO	station blackout
SCD	Statistical Core Design
SE	safety evaluation
SFP	spent fuel pool
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SM	Main Steam Supply System
SNSWP	Standby Nuclear Service Water Pond

SRP	NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants
SRSS	square-root-of-the-sum-of-the-squares
SSCs	structures, systems, and components
SSF	Standby Shutdown Facility
SV	Main Steam Vent to Atmosphere
TS	Technical Specification
TSO	Transmission System Operator
UHS	Ultimate Heat Sink
UFSAR	updated final safety analysis report
USE	upper shelf energy
V	volt
WG	Radioactive Waste Management Systems: Waste Gas
WL	Liquid Waste Recycle
WM	Liquid Waste Monitor and Disposal
WS	Nuclear Solid Waste Disposal System

If you have any questions, please call me at 301-415-2901.

Sincerely,

/RA/

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Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 269 to NPF-9
2. Amendment No. 249 to NPF-17
3. Safety Evaluation

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NAME	JBoska	SFigueroa	TTate*	AKlein	UShoop*	JAndersen*	JThorp*
DATE	04/12/13	03/25/13	07/17/12	07/03/12	07/20/12	11/08/12	01/29/13
OFFICE	NRR/EMCB/ BC	NRR/EPNB/ BC	NRR/EPTB/ BC	NRR/ESGB/ BC	NRR/EVIB/ BC	NRR/SBPB/ BC	NRR/SCVB/ BC(A)
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DATE	07/16/12	04/16/13	07/17/12	11/08/12	09/28/12	07/23/12	07/25/12
OFFICE	NRR/SNPB/ BC	NRR/SRXB/ BC	NRR/STSB/ BC	OGC - NLO	NRR/LPL2-1/ BC	DORL/ DD	NRR/LPL2-1/ PM
NAME	AMendiola*	CJackson*	RElliott	JWachutka	RPascarelli	MEvans	JBoska
DATE	11/30/12	11/30/12	04/09/13	04/25/13	05/09/13	05/16/13	05/16/13

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