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

April 29, 2016

Mr. Kelvin Henderson
Site Vice President
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (CAC NOS. MF4526 AND MF4527)

Dear Mr. Henderson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 281 to Renewed Facility Operating License (RFOL) NPF-35 and Amendment No. 277 to RFOL NPF-52 for the Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2), respectively. The amendments consist of changes to the RFOLs and the Technical Specifications (TSs) in response to your application dated June 23, 2014, as supplemented by letters dated August 26, 2014; December 15, 2014; January 22, 2015; April 23, 2015; and November 16, 2015. The amendments revise the RFOLs and TSs to implement a measurement uncertainty recapture power uprate at Catawba 1. As noted in the application, although the MUR uprate was for Catawba 1, the amendment request was submitted for both units. This is because the TSs are common to both units.

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

If you have any questions regarding this matter, I may be reached at (301) 415-4090 or by e-mail at Jeffrey.Whited@nrc.gov.

Sincerely,

Jeffrey A. Whited, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 281 to RFOL NPF-35
2. Amendment No. 277 to RFOL NPF-52
3. Safety Evaluation

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
Renewed License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (CNS-1, the facility), Renewed Facility Operating License No. NPF-35, filed by Duke Energy Carolinas, LLC (the licensee), dated June 23, 2014, as supplemented by letters dated August 26, 2014; December 15, 2014; January 22, 2015; April 23, 2015; and November 16, 2015, 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 Paragraphs 2.C.(1) and 2.C.(2) of Renewed Facility Operating License No. NPF-35 are hereby amended to read as follows:

   (1) Maximum Power Level

   Duke Energy Carolinas, LLC is authorized to operate the facility at reactor core full steady state power level of 3469 megawatts thermal (100%) in accordance with the conditions specified herein.

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

   A. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

   B. Coincident with the implementation of this amendment, the licensee shall revise the CNS Updated Final Safety Analysis Report (UFSAR). The revision shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

   C. Coincident with the implementation of this amendment, the licensee will fulfill the Regulatory Commitments identified in Attachment 1 of its license amendment request dated June 23, 2014, as supplemented by the Regulatory Commitment stated in letter dated November 16, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Anne T. Boland, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed License No. NPF-35
and the Technical Specifications

Date of Issuance: April 29, 2016

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (CNS-2, the facility), Renewed Facility Operating License No. NPF-52, filed by Duke Energy Carolinas, LLC (the licensee), dated June 23, 2014, as supplemented by letters dated August 26, 2014; December 15, 2014; January 22, 2015; April 23, 2015; and November 16, 2015, 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

   A. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

   B. Coincident with the implementation of this amendment, the licensee shall revise the CNS Updated Final Safety Analysis Report (UFSAR). The revision shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

   C. Coincident with the implementation of this amendment, the licensee will fulfill the Regulatory Commitments identified in Attachment 1 of its license amendment request dated June 23, 2014, as supplemented by the Regulatory Commitment stated in letter dated November 16, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Anne T. Boland, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed License No. NPF-52
and the Technical Specifications

Date of Issuance: April 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NOS. 281 AND 277

RENEWED FACILITY OPERATING LICENSE NOS. NPF-35 AND NPF-52

DOCKET NOS. 50-413 AND 50-414

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-35, page 3	NPF-35, page 3
NPF-35, page 4	NPF-35, page 4
NPF-52, page 4	NPF-52, page 4
TS Pages	TS Pages
1.1-5	1.1-5
3.4.3-3	3.4.3-3
3.4.3-5	3.4.3-5
3.7.1-3	3.7.1-3

(1) Duke Energy Carolinas, LLC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in York County, South Carolina, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) North Carolina Electric Membership Corporation to possess the facility at the designated location in York County, South Carolina, in accordance with the procedures and limitations set forth in this renewed operating license;

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

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

(5) Duke Energy Carolinas, LLC, 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;

(6) Duke Energy Carolinas, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein, and;

(7) Duke Energy Carolinas, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2, and Oconee Nuclear Station, Units 1, 2 and 3.

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

Duke Energy Carolinas, LLC is authorized to operate the facility at reactor core full steady state power level of 3469 megawatts thermal (100%) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC 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 December 6, 2024, 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) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

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, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> 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.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplement wherein this renewed license condition is discussed.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC 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 February 24, 2026, 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) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

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, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> 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.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

1.1 Definitions (continued)

NOMINAL TRIP SETPOINT	The NOMINAL TRIP SETPOINT shall be the design value of a setpoint. The trip setpoint implemented in plant hardware may be less or more conservative than the NOMINAL TRIP SETPOINT by a calibration tolerance. Unless otherwise specified, if plant conditions warrant, the trip setpoint implemented in plant hardware may be set outside the NOMINAL TRIP SETPOINT calibration tolerance band as long as the trip setpoint is conservative with respect to the NOMINAL TRIP SETPOINT.
OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14 of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
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 3469 MWt (Unit 1) and 3411 MWt (Unit 2).

(continued)

MATERIALS PROPERTY BASIS
Limiting Material: Upper Shell Forging 06,
Intermediate Shell Forging 05, and Bottom Head Ring 03
Limiting ART at 30.7 EFPY: 1/4-T, 42°F
3/4-T, 31°F

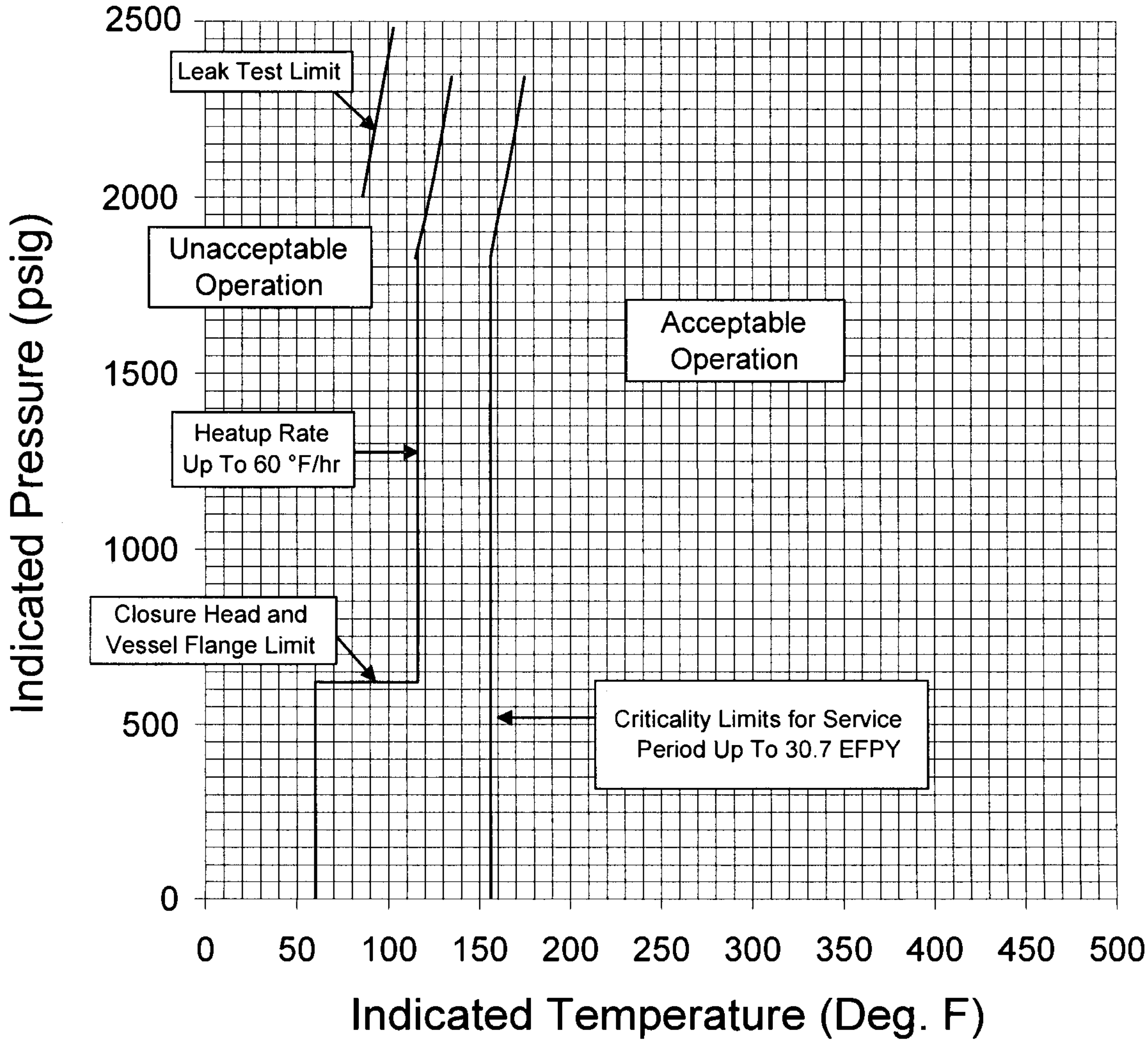

Figure 3.4.3-1
(UNIT 1 ONLY)
RCS Heatup Limitations

MATERIALS PROPERTY BASIS
Limiting Material: Upper Shell Forging 06,
Intermediate Shell Forging 05, and Bottom Head Ring 03
Limiting ART at 30.7 EFPY: 1/4-T, 42°F
3/4-T, 31°F

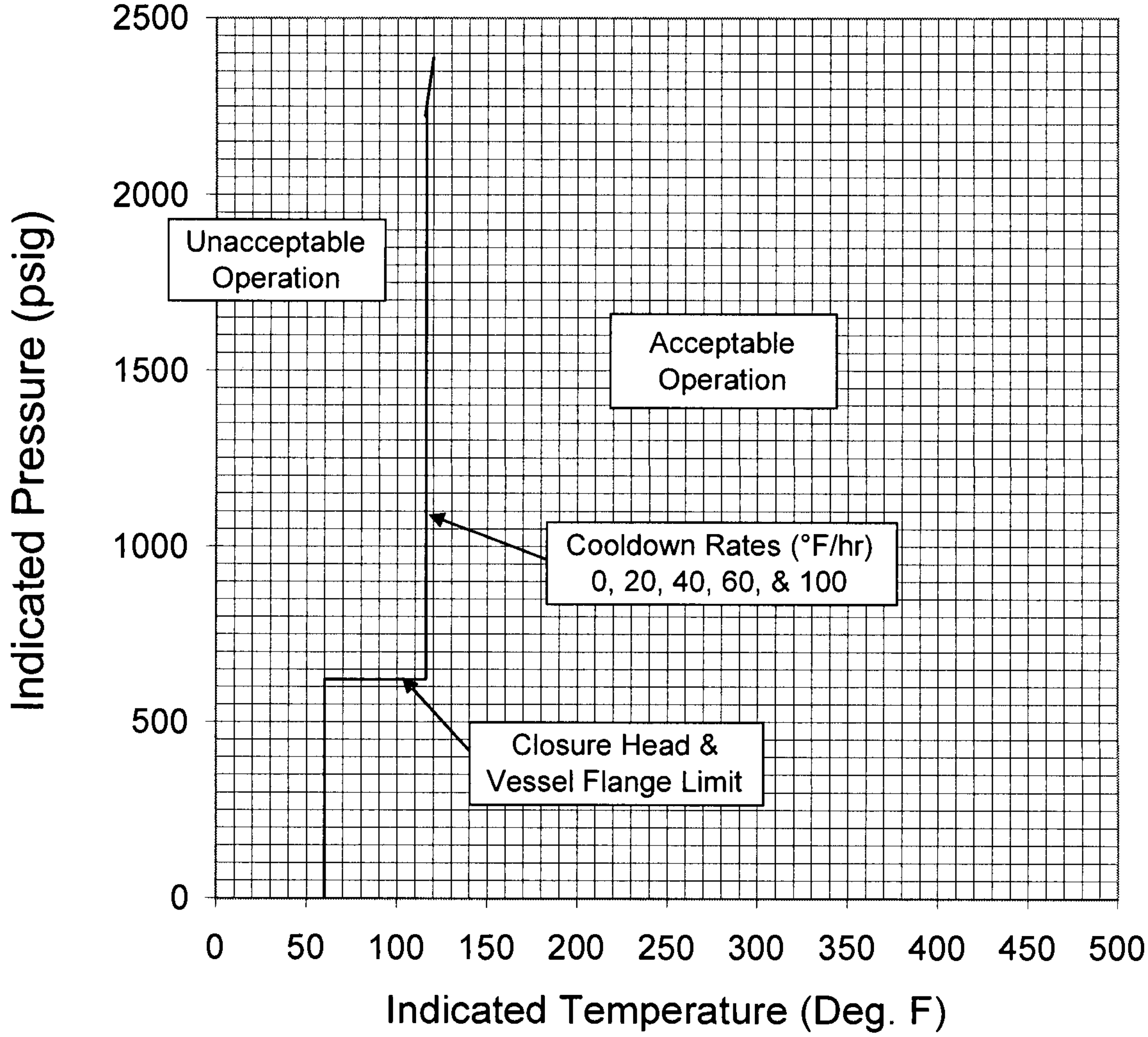

Figure 3.4.3-2
(UNIT 1 ONLY)
RCS Cooldown Limitations

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)	
	Unit 1	Unit 2
4	≤ 57	≤ 58
3	≤ 40	≤ 41
2	≤ 24	≤ 24

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	1175
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	1230

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-35

AND

AMENDMENT NO. 277 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By application dated June 23, 2014 (Reference 1, License Amendment Request (LAR)), as supplemented by letters dated August 26, 2014 (Reference 2); December 15, 2014 (Reference 3); January 22, 2015 (Reference 4); April 23, 2015 (Reference 5); and November 16, 2015 (Reference 6), Duke Energy Carolinas, LLC (Duke Energy, the licensee), requested changes to the Renewed Facility Operating Licenses (RFOLs) and the Technical Specifications (TSs) for the Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2).

Specifically, the licensee proposed changes that would revise the RFOLs and the TSs to implement a measurement uncertainty recapture (MUR) power uprate at Catawba 1. This amendment would raise the Catawba 1 rated thermal power (RTP) from 3411 megawatts-thermal (MWt) to 3469 MWt upon implementation, an increase of approximately 1.7 percent RTP. As noted in the application, although the MUR uprate is for Catawba 1, the amendment request was submitted for both units. This is because the TSs are common to both units.

The supplements dated August 26, 2014; December 15, 2014; January 22, 2015; April 23, 2015; and November 16, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 4, 2014 (79 FR 65429).

2.0 BACKGROUND

2.1 Measurement 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 in 10 CFR Part 50, Appendix K, was published in the *Federal Register* on June 1, 2000 (65 FR 34913), which became effective July 31, 2000. This change 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 an LAR, which must be reviewed and approved by the NRC staff. Catawba 1 is currently licensed to operate at a maximum power level of 3411 MWt, with a 2 percent margin in the ECCS evaluation model to allow for uncertainties in RTP measurement. The LAR would reduce this uncertainty to 0.3 percent.

In order to provide guidance to licensees seeking an MUR power uprate on the basis of improved FW flow measurement, the NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (Reference 7). RIS 2002-03 provides guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR power uprate LARs. While RIS 2002-03 does not constitute an NRC requirement, its use aids licensees in the preparation of their MUR power uprate LAR, while also providing guidance to the NRC staff for the conduct of its review. The licensee stated in its LAR that its submittal followed the guidance of RIS 2002-03.

2.2 Implementation of an MUR Power Uprate at Catawba 1

In existing nuclear power plants, the neutron flux instrumentation continuously indicates the RTP. This instrumentation must be periodically calibrated to accommodate the effects of fuel burnup, flux pattern changes, and instrumentation setpoint drift. The RTP generated by a nuclear power

plant is determined by steam plant calorimetry, which is the process of performing a heat balance around the nuclear steam supply system (called a calorimetric). The accuracy of this calculation depends primarily upon the accuracy of FW flow rate and FW net enthalpy measurements. As such, an accurate measurement of FW flow rate and temperature is necessary for an accurate calibration of the nuclear instrumentation. Of the two parameters, flow rate and temperature, the most important in terms of calibration sensitivity is the FW flow rate.

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. However, errors in the determination of flow rate can be introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter.[1] As a result of the desire to reduce flow instrumentation uncertainty to enable operation of the plant at a higher power while remaining bounded by the accident analyses, the industry assessed alternate flow rate measurement techniques and found that ultrasonic flow meters (UFMs) are a viable alternative. UFMs are based on computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling.

The licensee intends to use UFMs developed by the Cameron International Corporation (Cameron, formerly known as Caldon Ultrasonic Inc. (Caldon)), specifically the leading edge flow meter (LEFM) CheckPlus System, which provides a more accurate measurement of FW flow as compared to the accuracy of the venturi flow meter-based instrumentation originally installed at Catawba 1. 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, Engineering Report (ER)-80P, Revision (Rev.) 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System" (Reference 8), 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 for an exemption to the 2 percent uncertainty requirements in 10 CFR 50, Appendix K, in a safety evaluation (SE) dated March 8, 1999 (Reference 9), which allowed a 1 percent power uprate using the LEFM. Following the publication of the changes to 10 CFR 50, Appendix K, which allowed for an uncertainty less than 2 percent, Cameron submitted topical report ER-160P, Rev. 0, "Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM System" (Reference 10), a supplement to ER-80P. The NRC staff approved ER-160P by letter dated January 19, 2001 (Reference 11), for use in a power uprate of up to 1.4 percent at Watts Bar Nuclear Plant, Unit 1. Subsequently, in an SE dated December 20, 2001 (Reference 12), the NRC staff approved ER-157P, Rev. 5, "Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System" (Reference 13), for use in a power uprate of up to 1.7 percent using the CheckPlus system. By letter dated August 16, 2010 (Reference 14), the NRC staff approved ER-157P, Rev. 8, and

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

associated errata (References 15 and 16). ER-157P, Rev. 8, corrects minor errors in Rev. 5, provides clarifying text, and incorporates revised analyses of coherent noise, non-fluid delays, and transducer replacement. It also 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.

Catawba 1 was originally designed with FW flow and temperature instrumentation consisting of FW measurement nozzles, differential pressure transmitters, and thermocouples. Although the CheckPlus UFM system will be installed as part of the implementation of the MUR power uprate, 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.

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

As stated above, 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 its 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 RIS 2002-03.

3.1.1.2 Technical Evaluation

3.1.1.2.1 Leading Edge Flow Meter Technology and Measurement

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

The system uses multiple diagonal acoustic paths instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the FW mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the

derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM CheckPlus System in a plant-specific piping model at a calibration laboratory, Alden Research Laboratories (ARL).

The Cameron LEFM CheckPlus System uses 16 transducers, 8 each in two orthogonal planes of the spool piece. In the Cameron LEFM CheckPlus System, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Additionally, because there are twice as many acoustic paths in the CheckPlus System, than in the Check System, and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

3.1.1.2.2 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 MUR power uprate 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 MUR power uprate 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.

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

Items A and B request the licensee to identify and reference the documents that form the regulatory basis for the LAR. The licensee provided this information and specified that ER-80P and ER-157P form the regulatory basis for 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." The licensee identified Reference 9 and Reference 14 as the NRC staff SEs that approved the topical reports referenced.

The Cameron LEFM CheckPlus ultrasonic 8-path transit time flowmeter was installed at Catawba 1 in the spring of 2014. As discussed above, the CheckPlus design is described in Topical Reports ER-80P, ER-160P, and ER-157P, which 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 licensee indicated in the LAR that the LEFM CheckPlus ultrasonic flow meter system consists of an electronic cabinet and four measurement section/spool pieces (each consisting of eight electronic transmitters and eight pressure transmitters). One measurement section/spool piece

will be installed upstream of the FW control valves in each of the four 18 inch main FW flow headers. The measurement sections are located upstream of the existing FW flow venturis (two pressure transmitters and two CheckPlus transmitters per LEFM).

The licensee indicated in the LAR that the Catawba 1 LEFM CheckPlus system was calibrated and the FW piping configurations are explicitly modeled as part of the CheckPlus meter factor and accuracy assessment testing performed at ARL. The 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-996 (Reference 17) and ER-1009 (Reference 18), which are included in a proprietary attachment to the LAR.

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

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

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

Item D requests that licensees address 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, Rev. 8, in References 9 and 14, 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 Enclosure 2 to the LAR as well as in later supplements. The NRC staff evaluated the licensee's approach to addressing each of these criteria.

<u>Criterion 1 from ER-80P</u>

Criterion 1 requested a discussion of the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

The licensee stated that implementation of the MUR power uprate will include developing the necessary procedures and documents required for operation and maintenance at the uprated power level with the new LEFM CheckPlus system. A preventive maintenance program will be developed prior to implementing the LEFM CheckPlus system using Cameron's maintenance and troubleshooting manual and Duke Energy's established procedure program. The preventive maintenance activities include:

- General inspection of the terminal and cleanliness
- Power supply inspection of magnitude and noise
- Central processing unit inspection
- Analog input checks of the analog to digital (A/D) converter
- Watchdog timer checks that ensure the software is running
- Transducer cable checks of continuity and megger testing the cables
- Wall thickness check of each FW spool piece
- Calibration checks of each of the FW pressure transmitters
- Communication link checks

To address Criterion 1 from ER-80P, the licensee further stated, in Enclosure 2 to its LAR, in part, that:

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

Section 3.1.1.2.2.5 of this SE discusses contingency plans for plant operation with an inoperable LEFM. Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 1 from ER-80P.

Criterion 2 from ER-80P

Criterion 2 requests that plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in ER-80P.

In its LAR, the licensee stated that Criterion 2 does not apply to Catawba 1 as it did not have LEFMs installed at the time the LAR was submitted and was using flow venturis to measure FW flow to support the secondary calorimetric power measurements. Further, in its LAR the licensee provided the following regulatory commitment:

> After the LEFM CheckPlus system is installed and operational, thirty days of data will be collected comparing the LEFM CheckPlus operating data to the venturi data to verify consistency between the thermal power calculation based on the LEFM and other plant parameters.

LEFMs were installed at Catawba 1 during the spring of 2014. The licensee's regulatory commitment to verify consistency between the two flow rate measurements prior to implementation of the MUR power uprate, continues to satisfy this criterion. Further, as stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitment an implementation requirement of the MUR power uprate.

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

Criterion 3 from ER-80P

Criterion 3 requests that licensees confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current FW instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, then the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

The licensee stated that the LEFM uncertainty is based on the American Society of Mechanical Engineers (ASME) Performance Test Code 19.1, Instrument Society of America (ISA) Recommended Practice (RP) ISA RP 67.04, and ARL calibration tests. This methodology is consistent with the guidelines in NRC Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation" (Reference 19), ER-80P, and ER-157P.

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, which is discussed below in Section 3.1.1.2.2.3 of this SE. The uncertainty calculation was based on a square-root-sum-of-squares (SRSS) calculation. The LEFM uncertainty calculation method was provided in ER-1009, which was submitted with the licensee's LAR. The NRC staff reviewed this calculation, and determined that it is consistent with ER-80P and ER-157P. In addition, the licensee submitted ER-996 with its LAR, which the NRC staff reviewed and determined that it is consistent with the current heat balance uncertainty calculation that uses the FW flow nozzles and resistance temperature detectors.

The licensee's calculation for 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 NRC approved topical reports, and is consistent with the guidelines in RG 1.105.

Additionally, the LAR describes the licensee's commitment to perform acceptance testing following installation of the CheckPlus system in Catawba 1 to confirm that the built parameters are within the bounds of the error analyses, prior to implementation of the MUR power uprate, as discussed above. Also, in References 3 and 5, the licensee indicated that trend monitoring will be conducted to validate LEFM uncertainty. Additionally, the licensee stated that the LEFM FW pressure transmitters will be calibrated every 2 years to validate that the pressure measurement total uncertainty remains within the allowance documented in ER-996.

After reviewing the LAR the NRC staff determined that the methodology used to calculate uncertainty is based on accepted setpoint methodology and is consistent with the guidance in RG 1.105. Based on this review, the NRC staff concludes that the licensee has adequately addressed Criterion 3 from ER-80P.

Criterion 4 from ER-80P

Criterion 4 requests that for plant installation where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), the licensee should provide additional justification for use. This justification should show that the meter installation is either independent of the plant-specific flow profiles for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

To address Criterion 4 from ER-80P, the licensee stated, in Enclosure 2 to its LAR, in part, that:

> This criterion does not apply to Catawba, as the flow elements were tested and calibrated in a full-scale model of the Catawba Unit 1 hydraulic geometry at the [ARL]. A bounding calibration factor for the Catawba Unit 1 spool pieces was established by these tests and is included in the Cameron engineering report. An [ARL] data report for these tests and a Cameron engineering report (ER-1009 is 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 [LAR].

The NRC staff reviewed the information provided and concludes that the licensee adequately addressed Criterion 4 from ER-80P. Additionally, the LAR describes the licensee's commitment to perform acceptance testing following installation of the CheckPlus systems in Catawba 1, to confirm that the built parameters are within the bounds of the error analyses, prior to implementation of the MUR power uprate, as discussed above.

Criterion 1 from ER-157P, Rev. 8

Criterion 1 requests that licensees acceptably justify continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time on a plant-specific basis, with an in-operable LEFM.

The licensee's response and the NRC staff's review of this criterion are discussed below in Section 3.1.1.2.2.5 of this SE. Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 1 from ER-157P, Rev. 8.

Criterion 2 from ER-157P, Rev. 8

Criterion 2 states that, "A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty."

To address Criterion 2 from ER-157P, Rev. 8, the licensee stated, in Enclosure 2 to its LAR, in part, that:

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

Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 2 from ER-157P, Rev. 8.

Criterion 3 from ER-157P, Rev. 8

Criterion 3 states that an applicant with a comparable geometry can reference the finding in Section 3.2.1 of Reference 16 to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with the use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable ARL tests.

To address Criterion 3 from ER-157P, Rev. 8, the licensee stated, in Enclosure 2 to its LAR, in part, that:

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

Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 3 from ER-157P, Rev. 8.

Criterion 4 from ER-157P, Rev. 8

Criterion 4 states that an applicant requesting an MUR with the upstream flow straightener configuration discussed in Section 3.2.2 of Reference 16 should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 of Reference 16. Since the Reference 17 (from Reference 16) evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.

To address Criterion 3 from ER-157P, Rev. 8, the licensee stated, in Enclosure 2 to its LAR, in part, that:

> The existing feedwater flow venturis do not have a flow straightener. As discussed in Section I.1.C above, the feedwater flow venturis are located much greater than

4 L/D ([greater than] 200 feet) from the planned location of the LEFMs. The planned location of the LEFMs is also upstream of the feedwater flow venturis and will not include a flow straightener. Therefore, this criterion is not applicable to Catawba.

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, "An Evaluation of the Impact of 55 Tube Permutit Flow Conditioners on the Meter Factor of an LEFM CheckPlus" (Reference 20).

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 determined that it was 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 determined that the licensee has acceptably addressed the effects of flow straighteners. Based on this review, the NRC staff concludes that the licensee has adequately addressed Criterion 4 from ER-157P, Rev. 8.

<u>Criterion 5 from ER-157P, Rev. 8</u>

Criterion 5 requests that an applicant assuming large uncertainties in steam moisture content have an engineering basis for the distribution of the uncertainties or, alternatively, ensure that

their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 of Reference 16.

To address Criterion 5 from ER-157P, Rev. 8, the licensee stated, in Enclosure 2 to its LAR, in part, that:

> In 1996 and 1997, Duke Energy replaced the steam generators in Catawba Unit 1 and McGuire Units 1 and 2 with Babcock & Wilcox International (BWI) Model CFR-80 steam generators. The replacement steam generators were described in a BWI topical report that was attached to separate license amendment requests for Catawba Unit 1 and McGuire Units 1 and 2, both dated September 30, 1994. The Catawba Unit 1 steam generators were replaced in Fall 1996. ...Since Catawba Unit 1 was the lead unit for installation of the BWI Model CFR-80 steam generators, additional startup tests were performed, including moisture carryover testing. Moisture carryover testing on Catawba Unit 1 determined a moisture content of 0.051 +/- 0.006 [percent]. This test demonstrated the low moisture content from the BWI Model CFR-80 steam generators. These values were used as inputs in the calculation of the total power measurement uncertainty.

The NRC staff considers this uncertainty in steam moisture content to be small and not a significant factor in the calculation of the total power uncertainty 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. Based on this review, and the review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 5 from ER-157P, Rev. 8.

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

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 3.1.1.2.2.5 of this SE as noted above). The licensee stated that Criteria 2 and 4 from the NRC staff's SE for ER-80P, and Criteria 2 and 3 from the NRC staff's SE for ER-157P, Rev. 8, were not applicable to Catawba 1. The NRC staff reviewed these assessments by the licensee and determined that they are acceptable. The NRC staff reviewed the licensee's evaluation of Criteria 1 and 3 from the NRC staff's SE for ER-80P, and Criteria 4 and 5 from the NRC staff's SE for ER-157, Rev. 8, and determined that it is acceptable. Therefore, the NRC staff concludes that the licensee has adequately addressed the guidance in Item D of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

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

Item E requests that licensees submit a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contribution to the power uncertainty.

The licensee submitted ER-996 in Attachment 4 to its LAR to address Item E. In addition, the licensee provided Table I.1.E-1, which indicates that the uncertainty for the calorimetric inputs provided by the Cameron LEFM is 0.28 percent. The LEFM thermal power uncertainty was combined with the non-LEFM uncertainties to obtain a bounding total power uncertainty of 0.29 percent. These uncertainties were calculated using the calculation methodology described in ER-80P and ER-157P. The steam moisture content is considered when using the Topical Reports to establish uncertainties. A moisture content of 0.051 percent, plus or minus 0.006 percent, exists for Catawba 1, and these values were used as inputs in the calculation of the total power uncertainty.

The licensee's setpoint methodology approach statistically combined inputs to determine the overall uncertainty. Channel statistical allowances were calculated for the instrument channels. Dependent parameters are arithmetically combined to firm statistically independent groups, which are then combined using SRSS approach to determine the overall uncertainty. This methodology is consistent with the vendor's determination of the uncertainty of the Cameron LEFM CheckPlus System, as described in the referenced Topical Reports, and is consistent with the guidelines in RG 1.105.

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

The NRC staff reviewed the submittal and determined that the licensee properly identified the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power uncertainty.

The NRC staff has determined that the licensee has provided calculations of the total power measurement uncertainty for Catawba 1, and identified the parameters and their individual contributions to the overall thermal power uncertainty. Therefore, the NRC staff concludes that the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

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

Item F requests that licensees provide information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric (each aspect is followed by the licensee's response to address Item F as stated, in Enclosure 2 to its LAR):

> <u>Maintaining Calibration</u>
>
> RESPONSE:
> Calibration of the LEFM will be ensured by preventative maintenance activities previously described in Section [3.1.1.2.2.2 of this SE].

Controlling Software and Hardware Configuration

RESPONSE:
The Cameron LEFM CheckPlus Systems were procured to the requirements of ANSI/IEEE Std 7-4.3.2- 2003 (Reference I.6) and ASME NQA-1, 2008 (Reference I.7). Hardware configuration will be controlled in accordance with Duke Energy directive, NSD-301, "Engineering Change Program" (Reference [I].14).
LEFM software will be classified in accordance with Duke Energy directive EDM-809, "10 CFR 73.54 Critical Digital Asset Identification and Cyber Security Assessments" (Reference I.15). Software will be classified, developed, tested, and controlled in accordance with NSD-806, "Digital System Quality Program" (Reference I.17). Implementation of the software will be performed under the design control process governed by EDM-601, "Engineering Change Manual" (Reference I.18).

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

Performing Corrective Actions

RESPONSE:
Corrective actions will be monitored and performed in accordance with Duke Energy directive NSD-208, "Problem Investigation Program (PIP)" (Reference I.19).

Reporting Deficiencies to the Manufacturer

RESPONSE:
Reporting deficiencies to the manufacturer will be performed in accordance with Duke Energy directive NSD 208, “Problem Investigation Program (PIP)” (Reference I.19) and procurement specification.

Receiving and Addressing Manufacturer Deficiency Reports

RESPONSE:
Manufacturer deficiency reports will be received and addressed in accordance with Duke Energy directive NSD 208, “Problem Investigation Program (PIP)” (Reference I.19).

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

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

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

Items G and H request that licensees provide a proposed allowed outage time (AOT) for the instrument, along with the technical basis for the time selected, and to propose actions to reduce power if the AOT is exceeded.

To address Items G and H, the licensee stated, in Enclosure 2 to its LAR, in part, that:

> A Selected Licensee Commitment (SLC) will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is non-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 previously licensed rated thermal power). These SLC changes are not provided as part of this LAR but will be controlled using the 10 CFR 50.59 process.
>
> The basis for the proposed 72 hour [AOT] is as follows:
>
> - When an LEFM System is non-functional, signals from the existing feedwater flow venturis 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 flow venturi are calibrated to the LEFM signals, and upon LEFM failure, the flow venture calibration is locked to the last good LEFM value.
>
> - A statistical analysis and review of drift data for plant instrumentation providing the flow venture signals to the Secondary Calorimetric portion of the RTP calculation demonstrates that instrumentation and RTP drift should be insignificant over a 72 hour period. This indicates that, without application of a bias based upon a bounding value of RTP secondary calorimetric uncertainty, Catawba Unit 1 can be operated for 72 hours without exceeding the licensed RTP limit when the flow venturi signals are used as an input to the Secondary Calorimetric portion of the RTP calculation in place of the LEFM System.
>
> - A review of flow venturi fouling history demonstrates that fouling/de-fouling should not introduce significant error/drift over a 72 hour period. This indicates that, without application of a bias based upon a bounding value of RTP

secondary calorimetric uncertainty, Catawba Unit 1 can be operated for 72 hours without exceeding the licensed RTP limit when the flow venturi 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.

As stated above, the licensee provided the following regulatory commitment in its LAR:

> A Selected Licensee Commitment will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is non-functional.

The licensee stated that this regulatory commitment would be completed prior to the implementation of the MUR power uprate. As stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitment an implementation requirement of the MUR power uprate.

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

The NRC staff reviewed the information provided above for the proposed AOT. Based on its review, the NRC staff determined that the licensee provided sufficient justification for the proposed 72-hour AOT and the actions to reduce power level if the AOT is exceeded. Therefore, the NRC staff concludes that the licensee has provided the information requested by Items G and H of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

3.1.1.2.3 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.

3.1.1.2.4 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 UFM 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. Coverage of other aspects of the proposed use is addressed in other sections of this SE.

3.1.1.2.5 Test Facility Considerations

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

3.1.1.2.5.1 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 (Reference 21) was audited by the NRC staff in 2006 (Reference 22) 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 (References 23 and 24) and observed some improvements in test facility hardware. The NRC staff determined that these changes would not change its previous conclusions regarding test operations and results. In ER-1009, Cameron restated that "all elements of the lab measurements ... are traceable to NIST standards." Consequently, the NRC staff has determined 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 Catawba 1 CheckPlus spool pieces. An NRC staff audit confirmed that ARL was providing acceptable test data for the configurations under test. Consequently, the NRC staff has determined 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.

3.1.1.2.5.2 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. For an MUR power uprate LAR, licensees have 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 bounded 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.

Further, licensees have 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.

The licensee provided the test configurations as well as the in-plant CheckPlus locations in its LAR and ER-1009. As discussed below in Section 3.1.1.2.6.2 of this SE, distances between the exit of the CheckPlus spool pieces and the downstream flow straighteners and venturis are sufficient such that there will be no effect on the LEFM calibration. Test dimensions and configurations upstream of the LEFM were acceptable when compared to the plant installations. In addition, tests with offset orifices provided flow distortions that are judged to significantly bound any flow behavior that differs between the tests and plant.

The licensee stated that weigh tank tests were run at different flow rates for each simulated feedwater loop. Tests included a variation of flow rates through the CheckPlus and included an eccentric orifice upstream in the FW pipes containing the CheckPlus. Most test results were included in the reported main FW calculation.

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. Therefore, the NRC staff has determined that the licensee has acceptably addressed potential differences in testing configuration compared to the potential installation configuration.

3.1.1.2.6 In-Plant Installation and Operation of LEFMs

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

3.1.1.2.6.1 Transfer from Test to Plant and In-Plant Installation

For an MUR power uprate LAR, licensees 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. Further, the licensee must 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. ER-996 states that commissioning tests will be performed.

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.

The licensee addressed uncertainty at Catawba 1 in ER-996 and ER-1009 submitted with its LAR. As stated in SE Section 3.1.1.2.5.2, above, the uncertainty at Catawba 1 is acceptable. ER-996 provides transducer installation uncertainty. The content is essentially identical to Appendix D of Reference 16, which the NRC staff has previously determined is acceptable. Consequently, the Catawba 1 treatment of transducer installation uncertainty is acceptable. In its LAR the licensee provided the following regulatory commitment:

> After the LEFM CheckPlus system is installed and operational, thirty days of data will be collected comparing the LEFM CheckPlus operating data to the venturi data to verify consistency between the thermal power calculation based on the LEFM and other plant parameters.

As stated above, LEFMs were installed at Catawba 1 during the spring of 2014. Further, these parameters will be incorporated as required into the LEFM during commissioning. The NRC staff has reviewed this commitment and has determined that it is consistent with the approach the licensee has taken for transfer from test to plant and in-plant installation and is acceptable. Further, as stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitment an implementation requirement of the MUR power uprate.

3.1.1.2.6.2 In-Plant Operation

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, 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.D and I.1.F 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 the licensee's LAR and determined that the licensee's approach to in-plant operation is acceptable.

Operation with a failed LEFM CheckPlus system component was evaluated in Section 3.1.1.2.2.5 of this SE.

<u>Spool Piece Dimensional Effects on UFM Response</u>

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 determined that it is acceptable.

<u>Transducer Installation Sensitivity</u>

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. An uncertainty associated with the test loop would be expected 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 has determined 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. Reference 14 has established that licensees may use this methodology in their MUR power uprate LARs.

The LAR and ER-996 and ER-1009 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-996, the licensee reported test signal to noise 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 in the licensee's LAR and ER-996 and ER-1009. The NRC staff has determined that the licensee's approach for noise is sufficient to ensure that this topic is addressed acceptably.

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 3.1.1.2.5.2 of this SE above, separation from downstream components is needed so that CheckPlus operation will not be affected. Also as discussed above, the NRC staff determined that the in-plant separation from downstream piping components such as elbows and venturis is acceptable and will not affect CheckPlus operation. 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 the evaluation of the effect of downstream piping configurations on calibration and determined that it is acceptable.

Evaluation of Upstream Flow Straighteners on CheckPlus Calibration

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.

A previously undocumented effect of upstream tubular flow straighteners on CheckPlus calibration was discovered during ARL testing while NRC staff members were at the site on August 24, 2009, which did not appear to apply to any previous CheckPlus installations. As 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 ER-790P.

Cameron concluded in ER-790P 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.

Based on these insights, Cameron determined that the two uncertainty elements are uncorrelated and, therefore, combined them as the root sum squared to provide a quantitative uncertainty. The Cameron approach is judged to be valid, but there is concern that the characteristics of existing tubular flow straighteners in power plants may not be adequately represented by samples tested in the laboratory. Any licensee that requests an MUR power uprate with the configuration discussed above should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 20.

No flow straighteners are installed upstream of the LEFM locations in the Catawba 1 FW lines and flow straighteners are a significant distance downstream. Flow straightener effects are not a concern for this application. Accordingly, the NRC staff has reviewed the licensee's approach to evaluate the effect of upstream flow straighteners on CheckPlus calibration and determined that it is acceptable.

3.1.1.2.7 Other Thermal Power Calculation Considerations

3.1.1.2.7.1 Steam Moisture Content

Some modern separators and dryers deliver steam with moisture content in the 0.05 percent range and these applicants 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 no moisture is assumed.

ER-80P discusses an analysis in which the uncertainty in thermal power due to measurement of all variables excluding moisture is assumed to be normally distributed with two standard deviations of 0.3357 percent, essentially the aggregate uncertainty of all contributors excluding

moisture for the CheckPlus system. The contribution of uncertainty due to moisture content was then calculated by multiplying a second, uniformly distributed random number times the uncertainty band assumed in ER-80P, Table A-1 and Monte Carlo calculations of total power uncertainty were obtained. The results are summarized in Figure 1 of ER-80P. ER-80P, further states, in part, that:

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

This was stated to be an acceptable approach in Reference 16.

3.1.1.2.7.2 Deficiencies and Corrective Actions

In its LAR, the licensee identified its process for addressing Cameron deficiency reports as well as reporting deficiencies to the manufacturer. In each case Catawba 1 will use its corrective action program. In the case of receiving deficiency reports, Catawba 1 will document and address applicable deficiencies in its corrective action program as well. The NRC staff has determined that the licensee's response is acceptable.

3.1.1.2.7.3 Reactor Power Monitoring

As part of the MUR power uprate application, licensees should identify guidance to ensure that reactor thermal power licensing requirements are not exceeded. Proposed guidance was addressed by the NRC in letter dated October 8, 2008 (Reference 25).

During its review, the NRC staff requested that the licensee provide additional information discussing how Cameron Measurement Systems, ML205, Rev. 0, "Methodology for Calculating the Weighted Average of Several Measurements, Each Having an Estimated Uncertainty, to Minimize the Uncertainty of the Results" (Reference 26), will be implemented as part of procedures associated with the MUR power uprate.

The licensee responded to the NRC staff's request in Reference 3, stating, in part, that:

> Cameron document ML205 describes a methodology for identifying drift in baseline differences (biases) between independent parameters with a known relationship to feedwater mass flow rate. The ML205 method calculates a best-estimate feedwater mass flow rate by summing weighted diverse measurements. The difference between each diverse measurement and the best-estimate is then trended.
>
> Catawba's intention is to directly trend different measurements of plant power. This is considered to be equivalent to the ML205 method and allows direct comparison with additional diverse parameters (e.g., reactor coolant system delta-T and megawatt indicators). Also, comparison trending between venturi delta-P flow measurements and LEFM flow measurements will also be performed. Trend monitoring is not required to validate the LEFM calorimetric uncertainty;

> however, it is a prudent step that will be taken to further reduce the unlikely possibility of an overpower event.

The NRC staff has concluded that this response is acceptable based on the fact that the licensee has implemented ML205, Rev. 0, into the procedures and is trending an independent parameter, plant power, to feedwater mass flow rate.

3.1.1.2.8 Uncertainty

An uncertainty assessment is provided above in Section 3.1.1.2.2.2 of this SE. The following discussion provides supplemental information.
Cameron acceptably considers flow rate uncertainty associated with the test facility, measurement (including transducer installation), extrapolation from test conditions to plant operating conditions, modeling, and data scatter.

3.1.1.2.8.1 Test Facility Uncertainty

The budgeted test facility uncertainty is consistent with past NRC staff evaluations and the Reference 22 value. Therefore, this uncertainty is acceptable.

3.1.1.2.8.2 Measurement Uncertainty

In its LAR, the licensee addresses uncertainty due to such contributors as thermal expansion; dimensions; temperature, pressure, and density determination; and transducer installation. The contribution of some of these contributors was discussed above. Overall, measurement uncertainty is acceptably addressed.

During its review, the NRC staff requested the licensee to confirm that the Caldon Customer Information Bulletin (CIB) 119, Rev. 0, "Checklist Confirming the LEFM [Check] and LEFM [CheckPlus] Systems are Operating Within Design Basis" (Reference 27), will be followed while addressing the identified observation. The licensee responded to the NRC staff's request in Reference 3, stating, in part, that:

> Cameron/Caldon document Customer Information Bulletin CIB 119, Revision 0 identifies those parameters that must be monitored over time to ensure that the LEFM is operating within the bounds of its uncertainty analysis. Recent versions of the LEFM system (including the Catawba Unit 1 LEFM model) are designed to continuously self-monitor most of the parameters and conditions that require field verification. For the Catawba Unit 1 version of the LEFM CheckPlus system, Table 3 of CIB 119 identifies two parameters that require manual trending/adjustment as follows:
>
> - Periodic measurement of wall thickness using an ultrasonic thickness gauge, and
> - Periodic calibration of the feedwater pressure transmitters
>
> As part of the LEFM modification, the following changes to the plant's preventative maintenance program have been initiated to address the above:

1. The wall thickness of the LEFM spools will be measured using an ultrasonic thickness gauge (or equivalent instrument) once every refueling cycle to validate any change in internal diameter remains within the budgeted allowance of 0.015 inch documented within Cameron Engineering Report ER-996, Revision 1; and

2. The LEFM feedwater pressure transmitters will be calibrated every 2 years to validate the pressure measurement total uncertainty remains within the budgeted allowance of 15 psi documented within Cameron Engineering Report ER-996, Revision 1.

The NRC staff has determined that the licensee's response is acceptable based on the licensee following the guidance of CIB119 to monitor the two parameters: periodic measurement of wall thickness using an ultrasonic thickness gauge, and periodic calibration of the feedwater pressure transmitters that require manual trending.

3.1.1.2.8.3 Extrapolation Uncertainty

Although calibration tests were performed, they were conducted at room temperature. This resulted in Reynolds numbers about a factor of ten less than those are expected during operation in the plant, and an extrapolation is necessary to obtain in-plant calibration factor. A positive aspect of the LEFM CheckPlus is that the calibration factor is close to one and small errors in the extrapolation do not significantly affect extrapolation accuracy. Another aspect is that the LEFM Check and the LEFM CheckPlus characteristics permit an alternate extrapolation approach to the Reynolds number extrapolation. This involves the flatness ratio (FR), which for the LEFM CheckPlus is defined as the ratio of the average axial velocity (V) at the outside chords (chords 1, 4, 5, and 8) to the average axial velocity (V) at the inside chords (chords 2, 3, 6, and 7) as shown in the equation below:

$$\mathrm{FR} = (V1 + V4 + V5 + V8) / (V2 + V3 + V6 + V7)$$

Where FR is a function of Reynolds number, pipe wall roughness, and the piping system configuration.

The effect of the configuration is evaluated in laboratory tests. The effect of the Reynolds number is deduced from a flow profile correlation. The advantage of this approach is that a plot of FR versus calibration factor is linear and the calibration factor is insensitive to variation in FR. These results are consistent with analytic predictions and have been confirmed via ARL tests of many plant configurations. Further, minor changes in calibration factor observed in different hydraulics configurations are predictable and can be confirmed analytically. Therefore, if plant conditions result in a change in FR, the calibration factor may be adjusted to reflect the change in FR. This process is discussed further in Section 3.1.1.2.8.6 of this SE, below.

Cameron also uses swirl rate,[2] defined as:

$$\text{Swirl Rate} = \text{Average}\left[\frac{V_1 - V_5}{2 - y_S}, \frac{V_8 - V_4}{2 - y_S}, \frac{V_2 - V_6}{2 - y_L}, \frac{V_7 - V_3}{2 - y_L}\right]$$

Where y_S and y_L are normalized chord locations for outside/short and inside/long paths.

Cameron also uses swirl rate to characterize behavior obtained during ARL tests.

Catawba 1 provided experimental data of meter factor (MF) as a function of other parameters for each of the LEFM CheckPlus instruments in ER-1009. MF variation over the range of test flow rates was typically shown to vary by less than 0.001.

Cameron includes an uncertainty term for extrapolation from laboratory conditions to plant conditions that is computed from empirical equations to account for change in Reynolds number and other effects such as a difference in pipe wall roughness. The calibration factor is shown to change in the fifth significant figure over a factor of ten change in Reynolds number between the test and plant conditions. With respect to extrapolation uncertainty, some of the uncertainty was likely already addressed by parametric testing over Reynolds numbers and FRs.

The NRC staff has reviewed the licensee's method for determining extrapolation uncertainty and determined that it is acceptable.

3.1.1.2.8.4 Modeling Uncertainty

Cameron uses FR and swirl rate to characterize the velocity distribution and to validate the experimentally determined calibration factor when installed in a plant. Reference 29 includes a discussion of the application of calibration data obtained at ARL for 330 hydraulic configurations with 75 CheckPlus UFMs with an average calibration factor of 1.002 with a standard deviation of ±0.0039.

Cameron discussed its experience in calibrating over 100 UFMs with close to 500 different test configurations since typically 4 or 5 configurations were tested for each UFM. An approach is discussed where one configuration subset was considered applicable to the applicant's installation and modeling sensitivity was computed using that information.

The NRC staff has reviewed the licensee's method for determining modeling uncertainty and determined that it is acceptable.

3.1.1.2.8.5 Data Scatter Uncertainty

In its LAR, the licensee stated that the precision with which the calibration factor is determined includes calibration data for each LEFM CheckPlus. The licensee further stated that the

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[2] As shown in NRC letter dated June 18, 2004 (Reference 28), swirl can extend for significant distances and has the potential to affect calibration in some UFM designs.

95 percent confidence limits are calculated for test configurations that resemble the in-plant configuration.

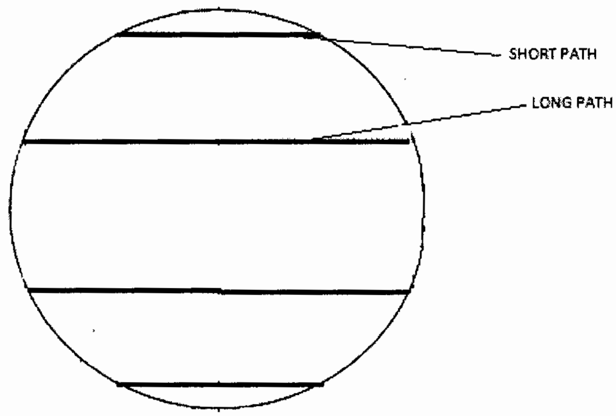
The NRC staff has reviewed the licensee's determination of data scatter uncertainty, and determined that it is acceptable.

3.1.1.2.8.6 Effect of flatness ratio change on meter factor

FR, as discussed above, is defined as:

FR = $(V1 + V4 + V5 + V8) / (V2 + V3 + V6 + V7) = V_S/V_L$

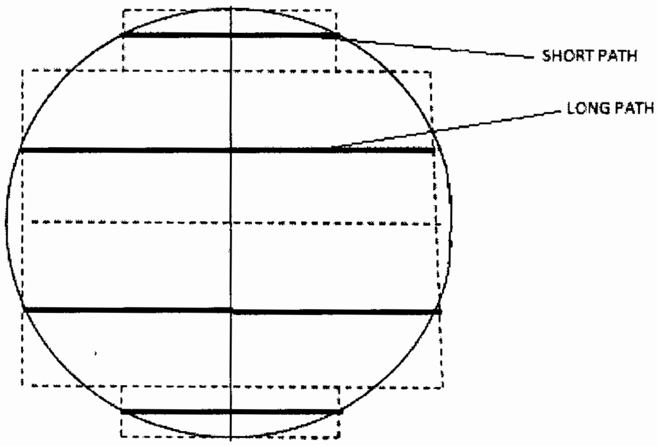
Where V_1, V_4, V_5, and V_8 are velocities measured along the outside chords (the short paths), V_2, V_3, V_6, and V_7 are velocities measured along the inside chords (the long paths), V_S is the mean short part velocity, and V_L is the mean long path velocity. The paths are illustrated by the horizontal lines in the following figure that correspond to the paths between the CheckPlus transducers:[3]



FR can be determined experimentally, such as by testing at ARL where the CheckPlus will provide the velocity data.

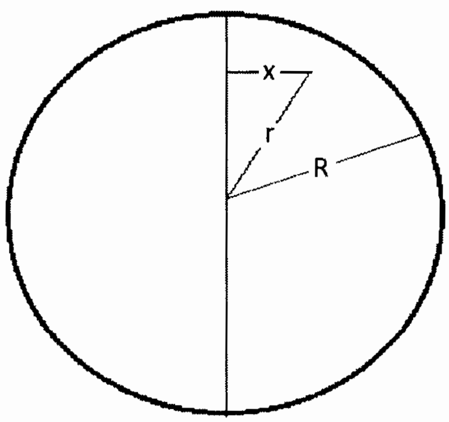
Once the Vs are determined, the flow rate determined by the CheckPlus can be calculated by multiplying the rectangular vertical widths (weighting factors) indicated in the following figure by the dash lines by the corresponding velocities times two:

[3] Measurements are at an angle with respect to pipe length. Velocities are translated into this configuration for calculation purposes.



Once the CheckPlus flow rate has been calculated, MF can be determined by comparing the CheckPlus flow rate to the experimentally determined data.

FR and MF can also be calculated using an assumed symmetric velocity distribution that is a function of pipe radius, expressed as V(r), where r is the reduced radial position with the origin at the pipe centerline and 0 less than or equal to r less than or equal to 1. Since the CheckPlus determines a mean velocity along the path, the calculation must be based on the same path, as illustrated by the "x" dimension in the following figure:



Where the mean velocity is calculated by the following equation:

$$1/X \int_0^X V(x)dx$$

Where x= X at r = R and V(x) is determined from the assumed V(r) where the relationship between x and r is obtained from the geometry illustrated in the figure.

The calculations define MF as the flow rate calculated by the following equation:

$$2\pi \int_0^1 V(r)\, r\, dr$$

Divided by the calculated LEFM flow rate obtained by two times the following equation:

$$\int V(x)\, dx$$

Over the short and long path lengths multiplied by the corresponding weighting factors. The calculations result in a linear relationship between MF and FR with little variation in MF. They further allow extrapolation of MF to the high Reynolds numbers in the plant that cannot be reached in the ARL tests by offsetting the calculated curve to pass through the data, which as shown in ER-1009, provide a good fit to the offset curve.

3.1.1.2.8.7 NRC Staff Conclusion Regarding Uncertainty

Based on the above evaluation, the NRC staff concludes that the licensee has adequately addressed the uncertainty considerations for flow rate uncertainty associated with the test facility, measurement (including transducer installation), extrapolation from test conditions to in-plant operating conditions, modeling, and data scatter.

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 power uprate.

The NRC staff has reviewed the licensee's response to RIS 2002-03, Attachment 1, Section I, and concludes that the licensee has met the guidelines contained therein. The NRC staff has determined that the licensee has adequately addressed the issues of FW flow measurement technique and power measurement uncertainty in its LAR. The NRC staff further concludes that 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 as well as other thermal power calculation considerations and uncertainty.

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 bases," states, in part, that, "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions

associated with ... postulated accidents, including loss-of-coolant accidents." The NRC staff reviewed the licensee's prediction of conditions in containment during postulated accidents.

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

The regulation at 10 CFR Part 50, Appendix J, Option B, states that Pa is defined as "the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications." As discussed below in this SE, evaluating the "Short-Term and Long-Term LOCA Mass and Energy Release and Containment Analysis," the Pa values in the Catawba 1 TS Section 5.5.2, "Containment Leakage Rate Testing Program," remain bounded after implementation of the MUR power uprate, by the analysis in the Catawba Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.1.1, which evaluates a large break LOCA at 102 percent of 3411MWt (3479 MWt).

Review guidance in the area of containment safety analysis can be found in several sections of the Standard Review Plan (SRP) (Reference 30), including Section 6.2.1, "Containment Functional Design"; Section 6.2.1.1.B, "Ice Condenser Containments"; Section 6.2.1.2, "Subcompartment Analysis"; Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents"; and Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures."

3.1.2.2 Technical Evaluation

The NRC staff reviewed the following areas of containment design and analysis for the LAR: Short- and Long-term LOCA containment response analyses; containment response to a main steam line break (MSLB); LOCA at a low power and reduced containment temperature; and minimum containment backpressure analysis.

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

The short- and long-term LOCA peak containment pressure analysis is documented in Section 6.2.1.1.3.1 of the Catawba UFSAR. UFSAR Section 6.2.1.3.1 contains the short-term M&E data for a LOCA, and was used as input for the containment sub-compartment analysis. This analysis was performed using the Westinghouse SATAN-V code. MSLB peak containment temperature analysis is documented in UFSAR Section 6.2.1.1.3.3. These analyses were performed to demonstrate that peak containment pressures and temperatures are acceptable and to ensure that the pressure and temperature profiles assumed in the EQ analyses were acceptable. The analyses described in the Catawba UFSAR were evaluated at 102 percent RTP (3479 MWt) and were reviewed and approved by the NRC.

Catawba UFSAR Section 6.2.1.1.3.2 describes the LOCA analysis at low power and reduced containment temperature. The analysis utilizes the Westinghouse Long Term Ice Condenser Containment Code LOTIC (References 31 and 32), which was used to reanalyze the long term response presented in UFSAR Section 6.2.1.1.3.1. The initial power used in the analysis is 5 percent and is, therefore, unaffected by the MUR power uprate.

Catawba UFSAR Section 6.2.1.5 describes the minimum containment pressure analysis, which provides the containment backpressure as an input to the LOCA peak clad temperature (PCT) analysis. The minimum containment backpressure is bounding for LOCA since it maximizes coolant loss and maximizes PCT. The power level used in the containment backpressure analysis is consistent with that assumed in the LOCA analysis. Lower power levels are conservative for the containment backpressure input to the ECCS evaluations; therefore, the current minimum containment backpressure analysis supports operation up to 102 percent RTP (3479 MWt).

The NRC staff reviewed the LOCA response analyses mentioned above and confirmed that the analyses were performed at 102 percent RTP (with the exception of LOCA at low power and reduced containment temperature, which is evaluated at 5 percent power) and remains bounding. Since the current analyses remain bounding, the Pa, in TS Section 5.5.2, remains unchanged for the MUR power uprate. Therefore, the NRC staff finds that TS 5.5.2 and the applicable Catawba procedures developed to address 10 CFR 50, Appendix J, remain acceptable at MUR power uprate conditions.

3.1.2.2.2 Postulated Secondary System Pipe Rupture Outside Containment

The NRC staff reviewed the secondary system pipe rupture M&E release effects to ensure that the doghouse equipment qualification temperature limit is not exceeded. By letter dated March 15, 1996 (Reference 33) the licensee submitted a response to an NRC staff information request, requesting to use RETRAN to calculate the M&E release outside containment per the NRC-approved methodology given in DPC-NE-3004-PA, Rev. 1, McGuire and Catawba M&E Release and Containment Response Methodology (Reference 34). The NRC staff accepted this response in a licensee amendment issued by letter dated August 29, 1996 (Reference 35). The RETRAN M&E release analysis is performed at an initial power level of 102 percent RTP

(3479 MWt) and acceptable temperatures were shown. Therefore, the analysis bounds the MUR power uprate proposed power level of 3469 MWt, and the NRC staff has determined that the analysis is unaffected by the LAR.

3.1.2.2.3 Fuel Handling Accidents in the Containment

With respect to fuel handling accidents in the containment, the accident analysis is performed to demonstrate that the offsite and control room doses are within regulatory limits. The accident analysis in Section 15.7.4.2.2, of the Catawba UFSAR was performed using the NRC reviewed and approved Alternative Source Term (AST) methodology in RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Reference 36). Since the source term was calculated at an initial power level of 102 percent RTP (3479 MWt), which bounds the MUR power uprate proposed power level of 3469 MWt, the NRC staff has determined that the analysis is unaffected by the LAR.

3.1.2.2.4 Environmental Qualification

With respect to EQ, the licensee states in part in Section II.1.D.iii Item 44 of Enclosure 2 of the LAR that:

> 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.

Temperature, pressure, and radiation conditions in the containment following a Large Break LOCA, MSLB, or fuel handling accident were discussed above. As noted, the Catawba UFSAR, analyses for these events were performed at 102 percent RTP (3479 MWt) and bound the MUR power uprate proposed power level of 3469 MWt. Therefore, the NRC staff has determined that there is no EQ impact with respect to temperature, pressure, or radiation due to the MUR power uprate. The NRC staff concludes that the EQ profile is conservative and acceptable with respect to operation at the proposed MUR power uprate power level of 3469 MWt.

3.1.2.2.5 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 Valve Injection Water System, 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 implementation of the MUR power uprate.

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

3.1.2.2.6 Containment Leakage Rate Testing Program

The Catawba Containment Leak Rate Test Program (CLRTP) is described in TS Section 5.5.2, "Containment Leakage Rate Testing Program." In Section VII.6.E of Enclosure 2 of its LAR, the licensee stated, in part, that:

> The Containment Leakage Rate Testing Program is discussed in Catawba Technical Specification Section 5.5.2. The MUR power uprate does not have any impact on the programmatic aspects of the Appendix J Program. It does not change any of the regulatory requirements of the program or change the scope of the program. The MUR power uprate does not change containment peak pressure following a large break LOCA since the UFSAR Section 6.2.1.1.1 assumed an initial power level of 102 [percent] of 3411 MWt (3479 MWt)...

The NRC staff has reviewed this response and determined that it is acceptable.

3.1.2.3 NRC Staff Conclusion Regarding Containment Systems

The NRC staff has determined that the current containment analyses remain bounding for the MUR power uprate described in the LAR. The NRC staff has also determined that 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. Further, the NRC staff, using the available SRP guidance, has determined that the criteria identified in GDC 4, GDC 16, GDC 38, and GDC 50 remain satisfied at MUR power uprate conditions. Therefore, the NRC staff concludes that the LAR is acceptable regarding Containment Systems.

3.1.3 Engineered Safety Features Heating, Ventilation and Air Conditioning Systems

3.1.3.1 Regulatory Evaluation

For Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning (HVAC) Systems, the NRC'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," is described in Section 3.1.2.1 of this SE. The effects of the release of post-accident fission products and toxic gases would be a consideration when evaluating Catawba 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"; Section 6.5.2, "Containment Spray as a Fission Product Cleanup System"; Section 9.4.1, Control Room Area Ventilation System"; Section 9.4.2, "Spent Fuel Pool Area Ventilation System"; Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System"; Section 9.4.4, "Turbine Area Ventilation System"; and Section 9.4.5, "Engineered Safety Feature Ventilation System."

3.1.3.2 Technical Evaluation

The NRC staff reviewed the impact of implementation of the MUR power uprate 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 of Enclosure 2 to the LAR, the licensee states that the control area ventilation system is designed to maintain the environment in the Control Room, Control Room Area and Switchgear Rooms within acceptable limits for the operation of unit controls, for maintenance and testing of the controls as required, and for uninterrupted safe occupancy of the control room during post-accident shutdown. The licensee also provided the following regulatory commitment in its LAR:

> A modification is planned to extend the outside air intakes for the Control Room and Control Room Area Pressurizing Subsystem to ensure the distance between the source-receptor pair is separated by 10 meters. This modification will ensure the design margin is maintained in the MSLB radiological dose calculation for the control room [Total Effective Dose Equivalent] TEDE. The MUR power uprate will not impact this modification.

The licensee stated that the modification is required to be complete prior to the implementation of the MUR power uprate. Further, as stated in Section 4.0 of this SE, the NRC staff has made

completion of the above regulatory commitment an implementation requirement of the MUR power uprate. The modification does not change the design basis (102 percent RTP) and, therefore, remains bounded for the MUR power uprate.

The licensee also stated that the auxiliary building ventilation system and the diesel building ventilation system remain bounded for the design basis (102 percent RTP) for the MUR power uprate conditions and that system design parameters are within the limits for all system components.

Additionally, the licensee stated that the containment purge and ventilation system is isolated and sealed during operation in Modes 1 through 4 and is not put into operation until the unit is in Mode 5; therefore, the functions of the system are not affected by the proposed 1.7 percent thermal power uprate.

3.1.3.3 NRC Staff Conclusion Regarding ESF HVAC

The NRC staff has determined 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, the NRC staff has determined that the criteria identified in GDC 4, GDC 19, GDC 41, GDC 60, GDC 61 and GDC 64 remain satisfied at MUR power uprate conditions. The NRC staff has also determined that applicable guidance in the SRP for evaluating the increase in heat loads in the control room and on the ESF ventilation systems has been adequately addressed. Therefore, the NRC staff concludes that the LAR is acceptable regarding ESF HVAC.

3.1.4 Plant Systems

3.1.4.1 Regulatory Evaluation

The NRC staff's review focused on verifying that the licensee has provided reasonable assurance that plant systems will continue to operate safely at the MUR power uprate conditions. 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 proposed MUR power uprate 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 proposed MUR power uprate on the plant systems in Enclosure 2 of the LAR. The NRC staff review below covers the impact of the MUR power uprate on the following major plant systems:

- NSSS interface systems,
- safety-related cooling water systems,
- SFP cooling analyses and systems,

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

The NRC staff's review concerning containment and ESF HVAC systems, which are also listed in RIS 2002-03, Attachment 1, Section VI, can be found in Sections 3.1.2 and 3.1.3, of this SE, respectively. The NRC staff conducted its review to verify that the licensee's analyses bound the proposed plant operation at the proposed MUR power level of 3469 MWt, and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate.

3.1.4.2 Technical Evaluation

3.1.4.2.1 NSSS Interface systems

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

3.1.4.2.1.1 Main Steam Supply System

The MSSS is described in Section 10.3 of the Catawba UFSAR. The MSSS 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 MSSS, 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 MSSS, 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 stated, in part, that:

> The review of the Main Steam System for the MUR power uprate shows that all system functions will continue to be performed following the MUR power uprate. The MUR power uprate conditions remain bounded by design as described in the Catawba UFSAR.

3.1.4.2.1.2 Condensate and Feedwater System

The Condensate and FW Systems are described in Section 10.4.7 of the Catawba 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

MUR power uprate condition (3469 MWt) and the current operating condition (3411 MWt). The comparison demonstrated that the condensate and FW systems have sufficient design and operational margin to accommodate the MUR uprate. The licensee determined that the proposed MUR uprate conditions remain bounded by design as described in the UFSAR.

3.1.4.2.1.3 Auxiliary Feedwater System

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 RTP (3479 MWt). The licensee stated that the analyzed core power level remains conservative and bounds the MUR power uprate (3469 MWt). Further, the licensee stated that AFW system maximum operating temperature and pressure remain essentially unchanged. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design or operation. There is no design change required for this system to operate at 3469 MWt. Therefore, the licensee stated that the AFW system is capable of supporting the proposed MUR power uprate.

3.1.4.2.1.4 NRC Staff Conclusion Regarding the NSSS System

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

3.1.4.2.2 Safety-Related Cooling Water Systems

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

3.1.4.2.2.1 Component Cooling System

The component cooling system is described in Section 9.2.2 of the Catawba 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 determined that the component cooling system will perform acceptably upon implementation of the MUR power uprate.

3.1.4.2.2.2 Nuclear Service Water System

The nuclear service water (NSW) system is described in Section 9.2.1 of the Catawba UFSAR. The NSW system 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.

The NRC staff has reviewed the licensee's analysis of the impact of the LAR on the NSW system and has determined that the NSW system will perform acceptably upon implementation of the MUR power uprate.

3.1.4.2.2.3 Ultimate Heat Sink

The UHS is described in Section 9.2.5 of the Catawba UFSAR. Two independent sources of NSW are available to provide a normal supply of cooling water: Lake Wylie 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 RTP (3479 MWt) for both units. This bounds the conditions after implementation of the 1.7 percent MUR power uprate (3469 MWt).

The NRC staff has reviewed the information above and determined that the UHS will perform acceptably upon implementation of the MUR power uprate.

3.1.4.2.2.4 NRC Staff Conclusion Regarding Safety Related Cooling Water Systems

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

3.1.4.2.3 Spent Fuel Pool 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 Catawba UFSAR states that the SFP 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 an MUR power uprate would be to the decay heat of the fuel recently discharged from the core. The licensee stated that the current analysis for SFP heat loads was performed at 102 percent RTP (3479 MWt).

The NRC staff does not expect that implementation of the proposed MUR power uprate will result in a significant change to the operation of the SFP storage and cooling system. Therefore, and

based on its review of the LAR, the NRC staff has determined that the SFP storage and cooling system will not be impacted by implementation of the MUR power uprate.

3.1.4.2.4 Radioactive Waste Management Systems

The Radioactive Waste Management Systems: Waste Gas (WG); Liquid Waste Recycle (WL); and 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 proposed MUR power uprate power level. The licensee stated that the radioactive waste management systems have no direct interface with the power cycle, and therefore, the MUR power uprate will have no impact on the ability of the systems to fulfill their functions. These systems are also credited for performing containment isolation for mitigating design basis events, which were analyzed at 102 percent RTP (3479 MWt).

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 has determined that the liquid and gaseous radioactive waste systems would perform acceptably after implementation of the MUR power uprate. 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. In its LAR the licensee stated, in part, that:

> The WS system has no direct interface with the power cycle, and therefore, the MUR power uprate will have no impact on this system.

The NRC staff has evaluated the licensee's analyses of the WS system and concludes that the WS would perform acceptably after implementation of the MUR power uprate.

3.1.4.2.5 Flooding Analyses

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, 7.6, 8.3, 9.2, 9.3, and 10.4 of the Catawba UFSAR. The licensee completed an engineering evaluation of the potential impact of the proposed MUR power uprate on internal flooding in the building and rooms currently discussed in the Catawba 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. Therefore, the licensee determined that the existing flood analyses remain valid and are not affected by operating at the increased power level described in the LAR.

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 concludes that the internal flooding analyses remain acceptable following implementation of the MUR power uprate.

3.1.4.2.6 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 pounds per square inch gauge (psig) and the maximum operating temperature equals or exceeds 200 degrees Fahrenheit (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 after implementation of the proposed MUR power uprate. No new lines are added, no break locations changed, and no change is made to the assumed blowdown from any postulated break. The licensee concludes that there is, therefore, no impact on the HELB analysis that was originally performed for Catawba 1.

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 concludes that the HELB analysis remains acceptable.

3.1.4.3 NRC Staff Conclusion Regarding Plant Systems Impacts

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

3.1.5 Accident Analyses

3.1.5.1 Regulatory Evaluation

In its LAR, the licensee generally concluded that existing analyses bounded the proposed MUR power uprate operating 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 RTP (3411 MWt). These analyses also bound plant operation at the proposed MUR power uprate power level of 3469 MWt, with an operating margin of 0.339 percent, which is greater than the stated 0.336-percent calorimetric power measurement uncertainty.
- For analyses that assume steady-state plant operation with a core power of 3411 MWt, the licensee evaluated the 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 below.

RIS 2002-03 states the following:

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

3.1.5.2 Technical Evaluation

The NRC staff utilized the approach discussed above in its review of the LAR. The NRC staff did not conduct a detailed review of the licensee's analyses that were performed at 102 percent RTP (3479 MWt). For these analyses, the NRC staff determined that existing analyses will continue to bound plant operation after implementation of the proposed MUR power uprate. Thus, the NRC staff has determined that these analyses are acceptable without a detailed review.

Table 3-1 below summarizes those areas of the accident and transient analyses that received a detailed NRC staff review, consistent with the guidance of RIS 2002-03.

Table 3-1 – Evaluation of Accident and Transient Analyses

Transient/Accident	**Analytic Power Level (percent RTP)**	**Review Comments**
FW System Malfunction that Results in a Reduction in FW Temperature	101.7	Acceptable
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 and 100	Acceptable
Steam System Piping Failure	0 and 100	Acceptable
Loss of External Load	102	Acceptable
Turbine Trip	102	Acceptable
Inadvertent Closure of Main Steam Isolation Valves	102	Acceptable
Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	102	Acceptable

Transient/Accident	Analytic Power Level (percent RTP)	Review Comments
Loss of Non-Emergency alternating current (AC) Power to the Station Auxiliaries	102	Acceptable
Loss of Normal FW Flow	101.7 Short-Term 102 Long-Term	Acceptable
FW System Pipe Break	101.7 Short-Term 102 Long-Term	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 (Locked Rotor)	101.7 and 102	Acceptable
Reactor Coolant Pump Shaft break	101.7 and 102	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	101.7	Acceptable
RCCA Misoperation (System Malfunction of Operator Error)	101.7	Acceptable
Startup of an Inactive RCP at an Incorrect Temperature	50	Acceptable
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	102	Acceptable
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	101.7	Acceptable
Spectrum of RCCA Ejection Accidents	0 or 102	Acceptable
Inadvertent Operation of ECCS During Power Operation	101.7	Acceptable
Inadvertent Opening of a Pressurizer Safety or Relief Valve	101.7	Acceptable
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	102	Acceptable
Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment	N/A	Acceptable

Transient/Accident	**Analytic Power Level (percent RTP)**	**Review Comments**
Steam Generator Tube Failure	101.7 DNB Analysis 102 SG overfill and T/H input	Acceptable
LOCAs	–	See discussion below
Radioactive Gas Waste System Leak or Failure	N/A	Acceptable
Radioactive Liquid Waste System Leak or Failure	N/A	Acceptable
Postulated Radioactive Releases Due to Liquid Tank Failures	N/A	Acceptable
Fuel Handling Accidents in the Containment and Spent Fuel Storage Buildings	102	Acceptable
Anticipated Transients Without Scram	102	Acceptable
Containment Performance Analysis	102	Acceptable
Postulated Secondary System Pipe Rupture Outside Containment	102	Acceptable
EQ Parameters	102	Acceptable
Flooding	101.7	Acceptable
Safe Shutdown Fire	102	Acceptable
Spent Fuel Pool Accidents	N/A	Acceptable

3.1.5.2.1 LOCA Analysis

The LOCA analyses currently in the UFSAR have been reviewed for the impact of a MUR power uprate. Specifically for the current best-estimate Large Break LOCA analyses, using 101 percent of RTP (3446 MWt) plus 1 percent uncertainty, the licensee determined that the PCT analysis is not bounded by the uprate. However, in its LAR the licensee made the following regulatory commitment, to be completed prior to implementation of the MUR power uprate:

> Duke Energy will re-evaluate the Loss-of-Coolant Accidents (UFSAR Section 15.6.5) consistent with the reload methodology.

This UFSAR update will include a PCT analysis performed at a best-estimate power of 101.7 percent of RTP (3469 MWt) with 0.3 percent uncertainty. As stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitment an implementation requirement of the MUR power uprate.

The PCT assessment for MUR conditions results in a PCT penalty of +16 F for the best-estimate Large Break LOCA analysis. The Small Break LOCA analysis is initiated from 102 percent RTP (3479 MWt), which bounds the proposed MUR power uprate power level of 3469 MWt including

uncertainty. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," contains five acceptance criteria: peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling that must be met for these analyses. The first four acceptance criteria are verified acceptable by providing the LOCA analyses, including the Large Break LOCA analysis performed at 101.7 percent of RTP (3469 MWt). The long-term core cooling is addressed in post-LOCA subcriticality, which is ensured during each reload core design.

The dose analysis was performed with a source term that assumes operation at 102 percent RTP (3479 MWt). The dose analysis utilizes the AST methodology, which was previously approved by the NRC staff.

In Reference 3, the licensee provided a clarification for the situation if the methodology used for PCT analysis for the MUR power uprate is equivalent to the current unbounded PCT analysis. The licensee additionally provided a justification for why the best estimate power levels of 101.7 percent RTP (3469 MWt) with 0.3 percent uncertainty is appropriate for the new PCT analysis compared to 101 percent RTP (3446 MWt) plus 1 percent uncertainty. The NRC staff reviewed the supplement information and determined that Westinghouse performed an evaluation for impacts to the Catawba 1 Large Break LOCA analysis, considering an MUR uprate to 101.7 percent RTP (3469 MWt). The assessment re-performs the Monte Carlo uncertainty analysis to quantify peak cladding temperature for the MUR power uprate. This is quantified by increasing the hot channel peaking factor (FΔH) and the total core peaking factor (FQ), which would correspond to the MUR rated thermal power, and also by reducing the assumed power uncertainty from 1.0 percent to 0.3 percent. The use of the Monte Carlo approach is consistent with the Code Qualification Document (CQD) Best-Estimate LOCA methodology, found in WCAP-12945-P-A, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models" (Reference 37), which is the Large Break LOCA evaluation methodology used in Catawba's current licensing basis.

The NRC staff has determined that this response is acceptable based on the Monte Carlo approach, found in WCAP-12945-P-A, which is consistent with the current licensing basis found in the Catawba 1 Core Operating Limits Report (COLR), submitted for Cycle 21 (Reference 38). In Reference 3, the licensee provided the PCT analysis performed at a best-estimate power of 101.7 percent RTP (3469) MWt with 0.3 percent uncertainty to confirm the accident and transient bounding for plant operation at the proposed uprated power level. The NRC staff reviewed the supplemental information provided with Reference 3 in the Westinghouse document DCP-05-14, Rev. 0, "Mini-Uprate (Appendix K Uprate) Evaluation of the Best-Estimate Large Break Loss of Coolant Accident for the McGuire, Unit 1 and 2, and Catawba, Unit 1 and 2, Nuclear Plants" (Reference 39). This evaluation shows that the new power level can be implemented without PCT margin recovery. The evaluation bounds the uprated power with the 0.3 percent uncertainty. The NRC staff concludes that this response is acceptable based on the PCT margin remaining adequate and the evaluation bounding the uprated power and 0.3 percent uncertainty.

3.1.5.3 NRC Staff Conclusion Regarding Accident Analyses

The NRC staff reviewed the current accident and transient analyses. Most of the current accident and transient analyses are based on operation at 102 percent RTP (3479 MWt) or 101.7 percent

RTP (3469 MWt) for departure from nucleate boiling (DNB) considerations. The LAR is based on the use of a Cameron LEFM CheckPlus 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 power uprate power level of 3469 MWt is bounded by the current accident and transient analyses and the NRC staff has determined that they are acceptable without performing a detailed review, consistent with the guidance contained in RIS 2002-03.

The NRC staff 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 criteria of 10 CFR 50.46 will continue to be met following a LOCA at the proposed MUR power uprate power level of 3469 MWt. The NRC staff reviewed the licensee's approach to reanalysis of the LOCAs and determined that it is acceptable. Therefore, the NRC staff has concluded that the LAR is acceptable regarding the changes to accident analyses.

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, 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 MUR power uprate, 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." As discussed in Section 3.2.6 of this SE, the neutron fluence values, which were considered as input parameters for PTS, P-T Limits, and USE evaluations are considered acceptable.

3.2.1.1 Pressurized Thermal Shock

3.2.1.1.1 Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of pressurized water reactor 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. The NRC staff's PTS assessment considered all beltline materials.

The determination of beltline as it applies to PTS assessment is explained as follows: Appendix H to 10 CFR Part 50 provides the requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline resulting from exposure to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 states that no material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that the peak neutron fluence at the end of the design life will not exceed 1×10^{17} neutrons/centimeter-squared (n/cm^2) with energy greater than one million electron volts ($E > 1$ MeV). Appendix G to 10 CFR Part 50 states, "To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of appendix H of this part." Furthermore, Section 2.2 of NUREG-1511, "Reactor Pressure Vessel Status Report" (Reference 40), states that the NRC staff considered materials with a projected neutron fluence of greater than $1.0E^{17}$ neutrons per square centimeter (n/cm^2) at end of license to experience sufficient neutron damage to be included in the beltline.

Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 ($E > 1$ MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period. This explanation is also summarized in RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 41).

3.2.1.1.2 Technical Evaluation

The licensee provided its PTS evaluation in Section IV.1.C.i of Enclosure 2 of its LAR, which states that PTS calculations were performed for Catawba 1 using the 60-year end-of-life extension (EOLE) neutron fluence values. The licensee further stated that Catawba 1 RV beltline materials will continue to meet the 10 CFR 50.61 PTS screening criteria.

The PTS screening criteria are 270 F for RV plates, forgings, and axial welds and 300 F for RV circumferential welds. For Catawba 1, the licensee stated that the limiting RT_{PTS} value was 63 F for the upper shell forging 06. However, in Table 5-42 of the Catawba UFSAR, the upper shell forging 06 is not mentioned. The NRC staff requested that the licensee provide additional information in order to confirm that the values of RT_{PTS} have been projected for all beltline materials, as defined in 10 CFR 50.61 and RIS 2014-11, and to clarify the apparent discrepancy between the LAR and the UFSAR regarding limiting material.

In Reference 4, the licensee verified that values of RT_{PTS} have been projected for all beltline materials expected to receive a neutron fluence of greater than 1×10^{17} n/cm^2. The licensee also provided fluence values for material adjacent to the beltline (such as inlet nozzles, outlet nozzles, and bottom head ring to lower vessel head circumferential weld) in order to demonstrate that the LAR addressed all beltline materials and that RT_{PTS} did not need to be projected for adjacent materials, since the adjacent materials were not expected to receive a neutron fluence of greater than 1×10^{17} n/cm^2. The licensee also addressed the apparent discrepancy between the LAR and the UFSAR by explaining that, as a result of the detailed plant-specific fluence analysis performed for the MUR power uprate, the nozzles no longer needed to be considered in the RT_{PTS}

evaluations. The licensee stated that the UFSAR will be updated to remove the out-of-date reference to the "bounding nozzle shell material."

After reviewing the licensee's response provided in Reference 4, the NRC staff confirmed that the values of RT_{PTS} have been projected for all beltline materials expected to receive a neutron fluence of greater than 1 x 10^{17} n/cm^2. The NRC staff has also determined that the licensee's proposal to remove "bounding nozzle shell material" from the UFSAR is acceptable. The NRC staff confirmed the upper shell forging 06 to be the limiting material for PTS. The NRC staff has also determined that the licensee's RT_{PTS} value of 63 F for the limiting upper shell forging 06 is consistent with the staff's own confirmatory calculations. Further, the NRC staff has determined that the MUR power uprate limiting RT_{PTS} value of 63 F is valid and does not exceed the PTS screening criteria.

3.2.1.1.3 NRC Staff Conclusion Regarding PTS

Since the RT_{PTS} values for the limiting RV beltline materials of Catawba 1 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 MUR power uprate, the Catawba 1 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

3.2.1.2.1 Regulatory Evaluation

Appendix G of 10 CFR Part 50 provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the reactor coolant pressure boundary (RCPB), including requirements 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 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 the proposed MUR power uprate.

3.2.1.2.2 Technical Evaluation

The licensee provided its P-T limit evaluation in Section IV.1.C.iii and its low temperature overpressure protection system (LTOPS) evaluation in Section IV.1.C.iv of Enclosure 2 of its LAR. The NRC staff determined that the current TS P-T Limits setpoints for Catawba 1 are based on one quarter of the RV wall thickness (¼T) adjusted reference temperature (ART) value at 34 EFPY of 42 F for the limiting material – the lower shell forging 04, and three quarters of the RV wall thickness (¾T) ART value at 34 EFPY of 31 F for the limiting material – the intermediate shell forging 05. The NRC staff has also determined that the LTOPS setpoints are established in conjunction with the P-T limit curves and are applicable for the same time period.

The licensee stated in its LAR that the limiting material for the location ¼T is changed from the lower shell forging 04 to the upper shell forging 06 and the ART at 34 EFPY is changed from 42 F

to 43 F. The licensee further stated in its LAR that the limiting material for the location ¾T remains the intermediate shell forging 05 (when using credible surveillance data) but the ART at 34 EFPY is changed from 31 F to 30 F. At the location ¾T, the bottom head ring 03 also shares the ART value of the limiting intermediate shell forging material. Because the limiting ¼T ART value (42 F) used in the development of the current P-T limit curves is slightly lower than the MUR power uprate limiting ¼T ART value (43 F), the MUR power uprate is not bounded by the current P-T limit curves at 34 EFPY. In order to evaluate when the limiting material at the ¼T location would have an associated ¼T ART value of 42 F using MUR power uprate fluence values, the licensee determined that the applicability date of the current P-T limits curves must be decreased from 34 EFPY to 30.7 EFPY. The table below summarizes the ART values and limiting materials for the ¼T and ¾T locations.

Table 3-2 – Summary of ART Values and Limiting Materials

	Current Requirement, 34 EFPY	**Proposed Requirement, 34 EFPY**	**Proposed Requirement, 30.7 EFPY**
Limiting Material, 1/4T	Lower shell forging 04	Upper shell forging 06	Upper shell forging 06
ART, 1/4T	42 F	43 F	42 F
Limiting Material, 3/4T	Intermediate shell forging 05	Intermediate shell forging 05/bottom head ring 03	Intermediate shell forging 05/bottom head ring 03
ART, 3/4T	31 F	30 F	Not specified, but less than 30 F

The licensee revised the applicability of the current TS figures from 34 EFPY to 30.7 EFPY and also revised the limiting material referenced on those figures. However, the P-T limit curves themselves were not redrawn.

Since, as discussed in RIS 2014-11, P-T limit calculations for ferritic RV materials other than those with the highest ART may define P-T curves that are more limiting because the consideration of stress levels from structural discontinuities may produce a lower allowable pressure, the NRC staff requested that the licensee provide additional information. Specifically, the NRC staff requested that the licensee describe how the current P-T limit curves consider all ferritic components in the RV that are predicted to receive a neutron fluence greater than 1 x 10^{17} n/cm^2 at the end of the licensed operating period.

In Reference 4, the licensee responded to the NRC staff's request, providing an analysis, which showed that the 4 inlet nozzles and 4 outlet nozzles would receive a neutron fluence at 34 EFPY of less than 1 x 10^{17} n/cm^2 at the end of the licensed operating period. The licensee's analysis for the nozzles included specific fluence values, with initial RT_{NDT}, chemistry factors, fluence factors, margin factors and ART. The licensee also included an evaluation of nozzle P-T limits that show the RV nozzle P-T limits to be bounded by the P-T limit curves provided in the LAR.

After reviewing the licensee's response, the NRC staff confirmed that the inlet and outlet nozzles would not receive a neutron fluence greater than 10^{17} n/cm^2 at the end of the licensed operating period. The NRC staff also confirmed, based on the response to the requested information that the P-T curves for the nozzles would not be more limiting than the P-T curves provided in the LAR. The NRC staff confirmed the analysis provided in the LAR, which shows the limiting material to be

the upper shell forging 06 for the ¼T location, and the intermediate shell forging 05 and bottom head ring 03 for the ¾T location. The NRC staff also performed confirmatory calculations, which support the revised applicability limit (30.7 EFPY) for the P-T curves as discussed in the LAR.

3.2.1.2.3 NRC Staff Conclusion Regarding P-T Limits

For the P-T limit evaluation, the NRC staff concludes that the RV beltline materials for Catawba 1 will continue to satisfy the P-T limit requirements specified in 10 CFR Part 50, Appendix G, after implementation of the MUR power uprate. The NRC staff also concludes that the revision of the applicability limit for the P-T limits in TS Figures to 30.7 EFPY is acceptable.

3.2.1.3 Upper Shelf Energy (USE)

3.2.1.3.1 Regulatory Evaluation

Appendix G of 10 CFR Part 50 provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RV materials against fracture. The NRC staff's review of the USE assessments covered the impact of the proposed MUR power uprate on the neutron fluence values and the USE values for the RV materials through the end of the current licensed operating period.

3.2.1.3.2 Technical Evaluation

The licensee provided the USE evaluation in Section IV.1.C.v of Enclosure 2 of its LAR and stated that the projected EOLE Charpy USE decreases due to MUR power uprate fluence at the ¼T location were calculated per RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 42). The licensee further stated that for Catawba 1, the limiting projected ¼T USE values are 60 ft-lbs for the bottom head ring 03. However, in Table 5-44 of the UFSAR, the bottom head ring 03 is not mentioned and the bounding nozzle shell material is listed with a projected EOLE Charpy USE of 50.1 ft-lbs. The NRC staff requested that the licensee provide additional information in order to confirm that the values of USE have been projected for all beltline materials, as defined in 10 CFR 50, Appendix G and RIS 2014-11, and to clarify the apparent discrepancy between the LAR and the UFSAR regarding limiting material.

In Reference 4, the licensee verified that values of USE have been projected for all beltline materials expected to receive a neutron fluence of greater than 1 x 10^{17} n/cm^2. The licensee also provided fluence values for material adjacent to the beltline (such as inlet nozzles, outlet nozzles, and bottom head ring to lower vessel head circumferential weld) in order to support the contention that the original LAR submittal addressed all beltline materials and that USE did not need to be projected for adjacent materials, since the adjacent materials were not expected to receive a neutron fluence of greater than 1 x 10^{17} n/cm^2. The licensee also addressed the apparent discrepancy between the LAR and the UFSAR by explaining that, as a result of the detailed plant-specific fluence analysis performed for the MUR uprate, the nozzles no longer needed to be considered in the USE evaluations. The licensee stated that the UFSAR will be updated to remove the out-of-date reference to the "bounding nozzle shell material."

After reviewing the licensee's response to the requested information, the NRC staff confirmed that the values of USE have been projected for all beltline materials expected to receive a neutron fluence of greater than 1 x 10^{17} n/cm^2. Although there was no data in the licensee's submittals to verify the validity of the USE value for "bounding nozzle shell material" in the UFSAR, the NRC staff determined that the drop in USE values due to irradiation are not required to be calculated for the nozzles since the nozzle materials are outside of the beltline region. Therefore, the NRC staff determined that the licensee's proposal to remove "bounding nozzle shell material" from the UFSAR is acceptable. Based on the NRC staff's review, the staff has determined that the bottom head ring 03 is the limiting material for USE, consistent with the licensee's statements in the LAR. The NRC staff has also determined that the licensee's USE value of 60 ft-lbs for the bottom head ring 03 is consistent with the staff's own confirmatory calculations. Further, the NRC staff concludes that the MUR power uprate limiting USE value of 60 ft-lbs is valid and exceeds the USE minimum requirements.

3.2.1.3.3 NRC Staff Conclusion Regarding USE

Since the EOLE USE value for all RV beltline materials of Catawba 1 will remain higher than the USE minimum requirement of 50 ft-lbs, after implementation of the proposed MUR power uprate, the NRC staff has concluded that the Catawba 1 RV beltline materials would continue to meet the USE minimum requirements of 10 CFR 50, Appendix G, and maintain adequate margin against fracture.

3.2.1.4 RV Material Surveillance Program

3.2.1.4.1 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. Appendix H of 10 CFR Part 50 provides the requirements for the design and implementation of the RV material surveillance program.

3.2.1.4.2 Technical Evaluation

In Section IV.1.C.vi of Enclosure 2 of its LAR, the licensee states, in part, that:

> The three required in-vessel surveillance capsules have been withdrawn and tested to date for Catawba Unit 1. The remaining capsules 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 Catawba Unit 1 reactor vessel, there is no longer a need to recommend a withdrawal schedule.

This information is consistent with that in the UFSAR, except that the licensee evaluation clarifies that of the three remaining capsules, which have been withdrawn without being tested, Capsule W, the sixth and last capsule to be withdrawn, was placed in the spent fuel pool following removal. The licensee's evaluation also clarifies that Capsule W was removed from the RV at 14.69 EFPY rather than at 13 EFPY and at a fluence of 3.51 x 10^{19} n/cm^2 rather than a fluence of

3.0×10^{19} n/cm^2. The NRC staff confirmed that Capsule W had accumulated sufficient neutron fluence to cover plant operation to 54 EFPY.

3.2.1.4.3 NRC Staff Conclusion Regarding the RV Material Surveillance Program

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 the licensee to provide surveillance capsule withdrawal schedules in the LAR for Catawba 1.

3.2.1.5 RV Internals and Core Support Structures

3.2.1.5.1 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 10 CFR 50 Appendix A, 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" (Reference 43), provides references to the NRC's approval of the recommended guidelines for RV internals in WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (Reference 44), and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (Reference 45).

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" (Reference 46), 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.

RIS 2002-03 contains guidance for licensee submittals of MUR power uprate LARs for uprates of power less than 2 percent RTP.

3.2.1.5.2 Technical Evaluation

The licensee discussed the impact of the LAR on the structural integrity of the RV internals in Section IV.1.A.ii of Enclosure 2 of its LAR. The licensee stated that the core delta temperature will experience a nominal increase of 1.7 percent, but that the revised core parameters are bounded by the design values plus uncertainty that were used in the current analyses. The licensee also stated that it has addressed the requirement of Section 7.2 of MRP-227-A and has implemented the elements identified in MRP-227-A, Section 7.3 in the Catawba 1 Inservice Inspection (ISI) Program.

The NRC staff reviewed the licensee's evaluation of the structural integrity of the Catawba 1 RV internals using the guidance of RIS 2002-03 and MRP-227-A. RIS 2002-03, Attachment 1, Section IV.1 requires that for components that are bounded by existing analyses of record (AOR), the discussion must include confirmatory information that explicitly states that the requested uprate in power level continues to be bounded by the existing AOR for the plant. Thus the licensee's discussion of reactor core support structures and vessel internals meets the guidance of RIS 2002-03.

Section 7.2 of MRP-227-A requires that each commercial pressurized water reactor unit shall develop and document a program for management of aging of reactor internal components within 36 months following issuance of MRP-227, Rev. 0 (that is, no later than December 31, 2011). In its LAR, the licensee stated that it had met this requirement by submitting a letter of intent dated June 16, 2010 (Reference 47) to adopt MRP-227 I&E guidelines for the Oconee Nuclear Station, Units 1, 2, and 3, McGuire Nuclear Station, Units 1 and 2, Catawba 1 and 2. Thus, the licensee submittal meets the requirements of Section 7.2 of MRP-227-A.

Section 7.3 of MRP-227-A requires implementation of the MRP-227-A aging management requirements, examination acceptance criteria, and expansion criteria within 24 months following issuance of MRP-227-A (that is, no later than December 31, 2013). In its LAR, the licensee documented that it has implemented the elements identified in MRP-227-A, Sections 7.3 in the Catawba 1 ISI Program. Thus, the licensee submittal meets the requirements of Section 7.3 of MRP-227-A. By letter of intent dated March 19, 2014 (Reference 48) the licensee stated its intent to submit plant specific reactor vessel internals inspection plans for Catawba 1 in the fall of 2022, which is at least 2 years prior to the initial inspection for reactor vessel internals. This is consistent with the guidance of NUREG-1801, Rev. 1, "Generic Aging Lessons Learned (GALL) Report" (Reference 49), which states that no further aging management review by the licensee is necessary if the licensee provides a commitment to:

(1) participate in the industry programs for investigating and managing aging effects on reactor internals;
(2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and
(3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

Based on the above, the NRC staff has determined that the licensee's management of the RV internals is consistent with the industry's I&E guidelines documented in MRP-227-A, and is, therefore, acceptable to the NRC staff.

3.2.1.5.3 NRC Staff Conclusion Regarding RV internals and Core Support Structures

The NRC staff has reviewed the licensee's evaluation of the impact of the MUR power uprate 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 MUR power uprate is acceptable because (1) the revised core parameters are bounded by the design values plus uncertainty that were used in the current analyses and (2) the licensee has addressed Section 7.2 of MRP-227-A

and has implemented the elements identified in MRP-227-A, Section 7.3 in the Catawba 1 ISI Program.

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

The NRC staff has reviewed the licensee's LAR and has evaluated the impact the proposed MUR power uprate will have on the structural integrity assessments for the RV and RV internals. The NRC staff has determined that the P-T limit applicability change from 34 EFPY to 30.7 EFPY is acceptable. The NRC staff has further determined that implementation of the MUR power uprate 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. Therefore, the NRC staff concludes that the LAR is acceptable regarding the RV integrity and RV internal and core support structures review.

3.2.2 Mechanical and Civil Engineering

3.2.2.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power, referred to as the RTP. 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 Catawba 1 will continue to be adequately maintained following the implementation of the proposed MUR power uprate under normal, upset, emergency and faulted loading conditions, as applicable. 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's 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"; 10 CFR 50, Appendix A, GDC 1, "Quality standards and records"; GDC 2, "Design bases for protection against natural phenomena"; GDC 4, "Environmental and dynamic effects design bases"; GDC 14, "Reactor coolant pressure boundary"; and GDC 15, "Reactor coolant system design."

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

The 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 MUR power uprate conditions. This, in turn, provides reasonable assurance that compliance with the applicable regulations will be maintained upon implementation of the proposed MUR power uprate. Section 3.1 of the Catawba UFSAR describes how the facility complies with the GDC.

The primary guidance used by licensees for MUR power uprate LARs is outlined in RIS 2002-03, which provides licensees with a guideline for organizing the LARs. 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.

3.2.2.2 Technical Evaluation

The NRC staff's review in the areas of civil and mechanical engineering covers the structural and pressure boundary integrity of the piping, components and supports, which make up the NSSS and the balance-of-plant (BOP) systems. The mechanical and civil engineering review scope also includes an evaluation of other new or existing SSCs, which are affected by the implementation of the proposed MUR power uprate. Specifically, this review focuses on the impact of the proposed MUR power uprate on the structural integrity of the Catawba 1 pressure-retaining components and their supports and the RV Internals. The NRC staff's review also 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 MUR power uprate on moderate energy pipe rupture locations was also performed. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and pressure boundary integrity of the aforementioned piping systems, components, component internals and their supports under

normal and transient loadings, including those due postulated accidents and natural phenomena, such as earthquakes.

The proposed MUR power uprate will increase the rated thermal power level from 3411 MWt to 3469 MWt at Catawba 1. In accordance with the 10 CFR 50, Appendix K requirements discussed above, the licensee notes in Section IV of Enclosure 2 of the LAR that the current ECCS analyses of record (AOR) are based on a core power level of 102 percent of RTP (3479). 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 RTP to a level lower than that for which the licensee has already analyzed.

3.2.2.2.1 Power Uprate Evaluation Parameters and Design Bases

In Table IV-1 in Section IV.1 of Enclosure 2 to the LAR the licensee provided the pertinent temperatures, pressures, and flow rates for the current and uprated conditions. The licensee evaluated the effects of the proposed MUR power uprate at a bounding power level of 102 percent RTP (3479 MWt). This power level corresponds to the proposed level following the implementation of the MUR power uprate (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 MUR power uprate. The RCS mechanical design flow of 147.8 million pounds mass per hour (Mlbm/hr) also remains unchanged after implementation of the MUR power uprate. At full power, the implementation of the MUR power uprate would yield a hot leg temperature of 614.9 F (from the current temperature of 614.4 F) and a cold leg temperature of 555.3 F (from the current temperature of 555.8 F), resulting in no change to the average RCS temperature. The main steam (MS) pressure decreases by 0.3 psia to 1020.7 psia at the MUR power uprate conditions and the MS steam flow increases from 15.1 Mlbm/hr to 15.5 Mlbm/hr at the MUR power uprate conditions. The FW temperature would increase by 2 F to 442 F as a result of MUR power uprate implementation.

The information related to the structural qualification of SSCs at Catawba 1 is contained in Chapter 3 of the UFSAR. The UFSAR describes the design criteria applicable to the Catawba 1 SSCs, including loads, load combinations, and acceptance criteria stipulated by the applicable codes of record for these SSCs. 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 proposed MUR power uprate has no effect on the deadweight and seismic loads of existing SSCs. Therefore, the NRC staff has determined that the loads used in the existing AORs for these SSCs remain valid.

The functional description of the RCS, including the RV, RCPs, RCS piping and SGs is discussed in Chapter 5 of the Catawba UFSAR. Chapter 10 of the Catawba UFSAR provides the design basis information for the secondary side systems, including the MS and the FW and condensate system. The licensee stated that all analyses and evaluations for SSCs performed to support this LAR, which are within the scope of the Catawba 1 license renewal effort were performed consistent with the methodologies outlined 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" (Reference 50).

3.2.2.2.2 Pressure-Retaining Components and Component Supports

As stated in Section IV.1 of RIS 2002-03, the LAR should contain

> A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing 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 MUR power uprate. If the AOR for a particular SSC was performed using conditions, which bound those that will be present at the proposed MUR power level, no further evaluation is required. The design codes of record for the Catawba 1 RCS are documented in Table IV.1.D-1 of Enclosure 2 to the LAR. The licensee confirmed that MUR power uprate evaluations did not include any changes to the tabulated design codes of record.

The pressure-retaining components and component supports, including piping and pipe supports, which must be evaluated in support of an MUR power uprate include the following: the reactor pressure vessel (RPV), including the RPV shell, RPV nozzles and supports; 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 that 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 proposed MUR power uprate. Affected components and supports refer

to those for which their AOR is not bounded at MUR power uprate conditions. Pressure-retaining components and their supports generally remain unaffected by the implementation of 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 MUR power uprate, identified above, affect the loads included in the AOR for the component and its supports. Based on its evaluations of the impact of MUR power uprate implementation on the components identified above, the licensee stated that the existing AORs related to the structural and mechanical qualifications of the following SSCs are unaffected by the proposed MUR power uprate at Catawba 1: the RPV, RPV nozzles and RPV supports; the pressure-retaining portions of the CRDMs; RCS piping and supports and loop branch nozzles; pressurizer shell, nozzles and surge line; the replacement SGs, including the shells, nozzles and secondary side internal support structures; and the pressure-retaining portions of the RCPs.

The NRC staff reviewed BOP piping as discussed in Section IV.1.A.v of Enclosure 2 of the LAR. The licensee's evaluation of the structural integrity of those BOP piping systems also demonstrated that the BOP piping systems will continue to meet their design basis under MUR power uprate conditions and remain bounded by the current AOR at MUR power uprate conditions. Similarly, the licensee confirmed that the MUR power uprate has no effect on the structural integrity of safety-related valves at Catawba 1 and these also remain bounded by their current AOR. Based on these considerations, the NRC staff concludes that all pressure-retaining components and supports, including piping and pipe supports, remain bounded at MUR 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, ensuring that there will be no impact on the structural and pressure boundary integrity of these SSCs at the MUR power uprate level; and (3) the magnitudes of plant parameter changes, as documented in Table IV-1 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 MUR power uprate.

3.2.2.2.3 RV Internals

In accordance with Section IV.1.A.ii of RIS 2002-03, the licensee evaluated the effects of the proposed MUR power uprate on the Catawba 1 RV internals. As discussed above, Section IV.1.B of RIS 2002-03 indicates that for those SSCs, including RV internals, whose AORs are affected by implementation of an MUR 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 MUR power uprate. The licensee summarized its evaluation of the effects of the proposed MUR power uprate on the structural integrity of the RV internals in Section IV.1.A.ii of Enclosure 2 to its LAR.

Mechanical and structural evaluations were performed by the licensee to determine any effects on the RV internals due to the conditions, which would be present following the implementation of the proposed MUR power uprate. The mechanical evaluations of FIV performed by the licensee are summarized in Section IV.1.B.iii of Enclosure 2 to the LAR. These evaluations focused on the potential for an increase in the vibratory response of the RV internals resulting from changes in the flow field at the MUR power level. An increase in vibratory response can introduce increased alternating stress intensities and subsequently higher cyclic fatigue of the RV internals. In Sections IV.1.B.iii of Enclosure 2 of the LAR the licensee stated, in part, 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 these considerations, the licensee confirmed that the FIV characteristics of the RV internals are bounded by the current AOR.

Additionally, in Sections IV.1.B.iv of Enclosure 2 of the LAR the licensee stated, in part, that:

> The changes in operating temperatures are provided in Table IV-1. The average temperature is unchanged, and the cold leg decreases 0.5 [F], while the hot leg temperature increases 0.5 [F]. These changes, as discussed elsewhere, have minimal impact on the MUR power uprate.

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

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

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

3.2.2.2.4 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 MUR power uprate. This assessment is summarized in Section IV.1.B.vii of Enclosure 2 to the LAR. The licensee stated in a summery to its assessment that the current AORs were reviewed to determine whether the MUR power uprate would have any impact on the current HELB AOR. The licensee 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 power uprate implementation. For the moderate energy line breaks (MELBs), the licensee also confirmed 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 power uprate implementation on jet impingement and thrust forces (dynamic effects) in Section IV.1.B.viii of Enclosure 2 to its LAR. The licensee stated that it had justified the elimination of large primary loop pipe rupture and pressurizer surge line pipe rupture from the design basis for Catawba 1 by using leak-before-break (LBB) concepts. The licensee also confirmed that piping loads used in the LBB evaluation are not affected by the power uprate and concluded that the LBB evaluation remains acceptable and is bounded by existing AOR. The licensee concluded that these are not affected by the implementation of the MUR power uprate due to the fact that the changes in the temperatures and pressures of these systems resulting from MUR power uprate 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 considers the licensee's assessments performed for these areas acceptable. This acceptance is based on the information presented above, which demonstrates that the AORs related to HELBs, MELBs and dynamic effects resulting from postulated pipe ruptures will remain bounding under the proposed MUR power uprate power level. The NRC staff considers this conclusion reasonable, given the small magnitude in temperature and pressure increases, which accompany MUR power uprate implementation. Correspondingly, as previously discussed, these small changes generally have no impact on pressure-retaining components such as piping.

3.2.2.3 NRC Staff Conclusion Regarding Mechanical and Civil Engineering

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on the structural and pressure boundary integrity of pressure-retaining components and supports and RV internals. Additionally, the NRC staff reviewed the licensee's assessment of the effects on the Catawba 1 HELB and MELB AORs, including associated dynamic effects. Based on the review above, the NRC staff concludes that the LAR is acceptable with respect to the

structural integrity of the aforementioned SSCs affected by the MUR 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 power uprate.

Specifically, the licensee demonstrated that: (1) the structural and pressure boundary integrity pressure retaining components and supports, including piping and pipe supports, at Catawba 1 are not affected by the proposed MUR power uprate, as evidenced by the fact that their AORs are unaffected; (2) the RV internals at Catawba 1 also remain unaffected, when considering the impact of MUR power uprate implementation on the FIV characteristics and structural integrity of the RV internals; and (3) the Catawba 1 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 Catawba 1 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.

3.2.3 Electrical Engineering

3.2.3.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. The regulatory requirements, which the NRC staff applied in this portion of its review include the following:

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.

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

10 CFR 50, Appendix A, 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.

3.2.3.2 Technical Evaluation

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

- AC Distribution System
- Power Block Equipment
- Direct Current (DC) System
- Emergency Diesel Generators (EDGs)

- Switchyard
- Grid Stability
- SBO
- EQ Program

3.2.3.2.1 AC Distribution System

The AC Distribution System is the source of power for the non-safety-related buses and for the safety-related emergency buses. According to Section 8.3 of the Catawba UFSAR, the AC sub-systems consist of the 13.8 kiloVolt (kV) normal electrical distribution system, 22kV and 230kV main power systems, and 4.16 kV and 6.9 kV normal and essential auxiliary electrical distribution systems.

The licensee stated in its LAR that Catawba 1 will see a load increase on the 6.9 kV buses due to the proposed MUR power uprate. However, this load increase is bounded by the current rated analysis and calculations of plant operations for these motors. The 6.9 kV loads that will be affected are two condensate hotwell pumps and two condensate booster pumps. The increase in these loads will not exceed the motor nameplate ratings and the 6.9 kV buses will have sufficient capacity.

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

3.2.3.2.2 Power Block Equipment

As a result of the proposed MUR power uprate, the RTP will increase from the previously analyzed core power level of 3411 MWt to 3469 MWt. The Catawba turbine-generator converts the thermal energy of steam produced in the steam generators into mechanical shaft power and then into electrical energy. Each unit at Catawba is operated primarily as a base loaded unit with an output of 1145 megawatt electric (MWe) net, but may be used for additional load when required. The generator is rated at 1450 mega-volt ampers (MVA) with 75 psig hydrogen pressure and a .90 power factor. Each Catawba unit produces a net electrical output of approximately 1145 MWe. At uprated conditions, the generator output for Catawba 1 will increase by 20 MWe. The licensee has stated that the increase in electrical output remains bounded by the design ratings of the generator. The licensee has performed reactive capability studies, which evaluated the capability of the generator and downstream components to generate or carry volt-amperes reactive (VARs). Increasing VARs are reflected in higher current being supplied to downstream components. The licensee submitted Figure V.1-1 "Reactive Operating Area," which demonstrates the relationship between the generator per unit (PU) voltage, generator VARs and switchyard PU voltage. On main step up transformer tap 3, the generator can provide leading and lagging VARs up to the generator capability limits.

Leading VARs are limited below generator rated voltage. With the proposed generating facility, the level of reactive support supplied by the addition to the unit has been determined to be

acceptable at this time. The study also concluded that the grid system reactive power capability is acceptable because adequate reactive support exists in the region.

The Catawba 1 main power system consists of the main generator, associated isolated phase bus, two generator circuit breakers, two half-sized unit step-up transformers (230/20.9kV), four half-sized unit auxiliary transformers (20.9/6.9/ 6.9kV), and one auxiliary transformer (22.8/13.8kV). In its LAR, the licensee states that the 22 kV main power systems, which include the isolated phase busses and generator circuit breakers will continue to have adequate capacity and capability for plant operation with an MUR power uprate. In addition, the licensee also reviewed calculations related to the generator circuit breakers and determined that the rating of each generator circuit breaker will remain bounded by the existing analysis and calculations for the MUR power uprate conditions. The unit step-up and unit auxiliary transformer calculations have also been reviewed by the licensee and were determined to be bounded by the existing analysis.

The NRC staff has determined that the generator is capable of operation at the proposed MUR power uprate conditions due to its rating meeting the grid MVAR requirements in relation to the increased generation on Catawba 1. In addition, the NRC staff has determined that each generator circuit breaker, isolated phase bus, and unit step-up and unit auxiliary transformers remains adequately sized for the MUR power uprate conditions. Based on its review of the LAR, the NRC staff concludes 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.

3.2.3.2.3 DC System

Section 8.3.2 of the Catawba UFSAR states that 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. In its LAR, the licensee stated that the DC systems are bounded by the existing analysis and calculations of record for the plant.

The licensee further stated in its LAR that an additional load would be added to the Electrical Computer Support (ECS) System in powering the LEFM. The licensee stated that this added load is within the rating of the ECS System. The ECS and uninterruptible power supply (UPS) systems provide 120VAC battery backed power in the event that offsite power is lost or in the event of loss of the shared bus. The ECS and UPS systems provide 120 VAC to the Operator Aid Computer (OAC) and Control Infrastructure Components installed in the OAC Room and Control Room, which are non-safety related. The NRC staff requested that the licensee provide additional information on the impact of the effects of the ECS additional load on the associated safety-related or non-safety-related buses with an MUR power uprate. In Reference 4, the licensee provided its response, stating that a loss of power to the components in the OAC Room and in the Control Room does not affect control functions in the Digital Control System (DCS) nor has the ability to initiate a plant transient or accident. In addition, the licensee stated that calculations CNC-1381.06-00-0023, "U1/2, 125 VDC Auxiliary Control Power System Battery and Battery Charger Sizing," and CNC-1381.06-00-0071, "ECS System UPS 120V Load Calculation,"

have been revised for the MUR power uprate and it has determined that Catawba continues to have adequate capacity and capability for plant operations with an MUR power uprate.

The NRC staff reviewed the LAR, Reference 4, and the Catawba UFSAR and confirmed that the MUR power uprate would not impact any DC powered indication, control, or protection equipment. Therefore, the NRC staff concludes that the analyses for the DC system bound MUR power uprate conditions.

3.2.3.2.4 Emergency Diesel Generators

The standby emergency AC power source for each Catawba unit consists of two diesel generators. The EDG system automatically supplies emergency electrical power to the ESF plus selected BOP emergency loads in the event that the normal AC power is interrupted.
In its LAR, the licensee stated that there are no load changes to the emergency bus loads supported by the EDGs due to the proposed MUR power uprate, and that the existing accident analyses remain bounding. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR power uprate conditions.

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

3.2.3.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The switchyard serves six 230-kV primary transmission lines. The primary function of the switchyard and distribution system is to connect the station electrical system to the transmission grid. The current to the switchyard is bounded by the main transformers capability. The small increase in plant output after implementation of the MUR power uprate does not significantly impact the switchyard equipment.

Based on its review of the licensee's LAR, the NRC staff concludes that the analyses for the switchyard system at Catawba can reasonably bound the MUR power uprate conditions.

3.2.3.2.6 Grid Stability

In its grid stability impact of the proposed MUR power uprate as discussed in the LAR, the licensee concludes that there is no significant effect on grid stability or reliability. The grid stability impact study included a thermal analysis, fault duty, stability, and reactive capability study. Based on these studies, the licensee determined that there is no fault duty, stability, or interconnection impacts on the grid transmission system, and therefore, no additions or modifications are required to accommodate the proposed MUR power uprate.

After reviewing the LAR, the NRC staff requested that the licensee provide additional information on the grid stability study, specifically requesting that the licensee provide an explanation of the power flow analysis used in the generation impact study, which was modified to include 20 MWe of additional generation at Catawba 1.

The licensee responded to the NRC staff's request in Reference 4 stating, in part, that:

> No network upgrades were identified as being attributable to the studied generating facility. This determination is based on a comparison of a model of the transmission system as is and a model of the transmission system that includes the additional generating capacity at Catawba.
>
> Consistent with Duke Energy's generator interconnection studies and transmission planning practices, the power flow analysis was only performed on a summer model. Duke Energy is summer peaking, so the summer models contain higher loads. Summer models also contain lower facility ratings than winter models because they are based on higher ambient temperatures. The pairing of higher loads and lower ratings creates the most stressed condition on the transmission system.
>
> At the time the request was studied, higher queued generators were modeled. In all scenarios studied, the requested additional output at Catawba is modeled and the remaining Duke Energy generators are economically dispatched to account for the 20 MW increase at Catawba.

Based on its review of the LAR and Reference 4, the NRC staff has determined that the results of this study indicate that the increase in loading during MUR power uprate conditions will be encompassed by the grid and will not affect current or future plant outputs. Therefore, the NRC staff concludes that the Catawba MUR power uprate allows for continued stable and reliable grid operation.

3.2.3.2.7 Station Blackout

For Catawba 1, the SBO scenario assumes that both Catawba 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 Catawba 1 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 Nuclear Management and Resources Council (NUMARC) 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" (Reference 51) as discussed in Section 8.4.2 of the Catawba UFSAR. In Section V.1.B of Enclosure 2 of its LAR, the licensee stated that the alternate AC (AAC) power source provided at Catawba 1 is the Standby Shutdown Facility (SSF) diesel generator (DG). The licensee further stated that the SSF DG is available within 10 minutes of an SBO event and has sufficient capacity and capability to operate equipment necessary to maintain a safe shutdown condition for the four-hour SBO event.

The licensee stated that the evaluation for SBO included the adequacy of the AAC source, condensate storage tank inventory, Class 1E battery capacity, compressed air, containment isolation, and reactor coolant system inventory currently credited for SBO mitigation. The AAC source has sufficient capacity to operate systems necessary for coping with an SBO event for the

required coping period. The licensee also stated that the condensate storage tank inventory is adequate for decay heat removal following a SBO event at uprated conditions.

The licensee further stated that the proposed MUR power uprate has no effect on the Catawba 1 station battery capacity as the MUR power uprate does not increase loads. No air-operated valves are relied upon at Catawba 1 to cope with a four-hour SBO event. However, air can be supplied from a diesel-driven air compressor and/or an instrument air system compressor powered from the non-SBO unit. The licensee also stated that the backup air capability provides operators with flexibility to maintain hot standby conditions from the main control room. Power-Operated Relief Valves and auxiliary feedwater flow control valves can be manually operated to maintain hot standby conditions during the four hour SBO duration. Therefore, the ventilation for areas containing SBO equipment is unaffected by the MUR power uprate.

Based on its review of the licensee's LAR, the NRC staff has determined that the MUR power uprate will have no impact on the Catawba 1 SBO coping duration. Therefore, the NRC staff concludes that Catawba 1 will continue to meet the requirements of 10 CFR 50.63 upon implementation of the MUR power uprate.

3.2.3.2.8 Environmental Qualification Program

In Section V.1.C of Enclosure 2 of its LAR, the licensee stated that an evaluation of EQ parameters such as temperature, pressure, and radiation was performed to evaluate whether any potential parameter would change due to the proposed MUR power uprate. When evaluating the temperature and pressure parameters the BOP systems showed some slight parameter changes but these were shown to have no impact on the EQ components at Catawba 1. The evaluation of the systems inside Containment and the Doghouse buildings for accident temperature and pressure conditions showed that the current design basis analyses were performed at 102 percent RTP (3479 MWt), which bounds the MUR power uprate conditions. The ratings of these areas did not change from mild to harsh as a result of the MUR power uprate temperature conditions. Therefore, the licensee concluded that there is no EQ impact with respect to temperature or pressure parameters due to the MUR power uprate.

The licensee further stated that the Catawba 1 components listed on page E2-27 of its LAR that are located in the shared auxiliary building between both Catawba units have not yet been confirmed to be qualified at the MUR power uprate total integrated dose (TID). As a result of this, the licensee submitted the following regulatory commitments, commitments 9 and 10, in its LAR:

> Duke Energy will resolve the issue of the qualification of the six ITT Barton pressure transmitters in the Reactor Vessel Level Indication System that was identified by the EQ review for being qualified to the post-MUR power uprate TID.
>
> Duke Energy will resolve the issue of the qualification of the Struthers Dunn Type 219 relay that was identified by the EQ review for being qualified to the post-MUR power uprate TID.

After reviewing the LAR, the NRC staff requested that the licensee provide additional information on the radiation or environmental effects on these components. In Reference 4, the licensee

responded to the NRC staff's request by stating that it had been determined through further analysis that the upper bound TID adjusted for the MUR power uprate at the installed location of the six ITT Barton transmitters was below the tested qualification dose. The licensee also stated that further analysis of a detailed materials evaluation concluded that the Struthers Dunn Type 219 relay is qualified for MUR power uprate conditions. The licensee further stated that with this analysis and review, the six pressure transmitters and one Struthers Dunn Type 219 relay are determined to be qualified for MUR power uprate conditions. The licensee stated that this analysis completes LAR commitments 9 and 10.

The NRC staff also requested that the licensee provide additional information concerning the following regulatory commitment, commitment 14, provided with the LAR:

> Duke Energy will evaluate the 50 equipment IDs and the 40 enclosures containing 21 individual component types encompassing 330 total components for post-MUR EQ conditions.

At the time that the LAR was submitted these components had not yet been confirmed to be qualified at the MUR power uprate. However, in Reference 2, the licensee submitted additional EQ information that stated that these components had been qualified and were acceptable for post-MUR EQ conditions. The NRC staff requested that the licensee provide a summary of the results stating the basis of the EQ evaluations that were performed and any potential impacts the results may have on the proposed MUR power uprate. The licensee responded to the NRC staff's request in Reference 4, stating, in part, that:

> As discussed in LAR Section II.1.D.iii item 44 "Equipment Qualification (EQ) parameters," during the review of EQ documents it was discovered that 50 component IDs were left out of the original evaluation. These were 40 incore thermocouples, four potential transformers, four fuses, and two damper operators. Further review of the incore thermocouples determined that they contained no age-sensitive materials and were therefore not susceptible to radiation effects. Evaluation of the four potential transformers, four fuses, and two damper operators determined that the post-MUR TIDs experienced by these components were within their tested TID. As shown in Table 1, all 50 components are qualified for post-MUR conditions.
>
> While incorporating lessons learned from the McGuire MUR, Catawba identified a number of electrical enclosures that had not been evaluated on an individual component basis. Individual components in the electrical enclosures were identified and these components were reviewed using the same methodology as the other EQ components - by comparing the component's qualified radiation exposure to the TID adjusted for MUR. As shown in Table 2, all of these additional components were found to be qualified for post-MUR conditions.

Based on the review discussed above, the licensee considers LAR commitment 14 to be complete.

The licensee also determined that an EQ evaluation was needed for Radiation Zones 30 and 45 providing the following regulatory commitments, commitments 11 and 12, in its LAR:

> Duke Energy will resolve the issue of portions of one area (Radiation Zone 30 in the Catawba Auxiliary Building at the 577 foot elevation) that were found to exceed the normal operating 40-year dose listed in the Catawba Environmental Qualification Criteria Manual for pre-MUR power uprate conditions.
>
> Duke Energy will resolve the issue of portions of one area (Radiation Zone 45 in the Catawba Auxiliary Building at the 594 foot elevation) that were found to potentially exceed the normal operating 40-year dose listed in the Catawba Environmental Qualification Criteria Manual for post-MUR power uprate conditions.

In Reference 4, the licensee stated that the above commitments were still not completed. Subsequently, in Reference 5, the licensee stated, in part, that:

> In followup reviews associated with LAR Commitments 11 and 12, Duke Energy has continued evaluations of the radiological dose values associated with Radiation Zones 30 and 45 in the Auxiliary Building. This evaluation has recently identified existing legacy dose shielding discrepancies with several Annulus penetrations located at different elevations in the Auxiliary Building. Due to the potential increase in the calculated dose levels in these areas, Catawba Condition Report (CR) C-15-00304 was entered into the Corrective Action Program.
>
> The initial evaluation of CR C-15-00304 prompted entry into the Operability Determination Process for equipment in the Electrical Penetration and 4 KV Switchgear Rooms on Elevations 560[ft] and 577[ft] and the Electrical Penetration Rooms on Elevation 594[ft] for both [Catawba] Unit 1 and Unit 2. In total, seventeen (17) penetrations were identified with insufficient lead shielding, with eleven (11) on Unit 1 and six (6) on Unit 2. This resulted in elevated dose values calculated for the identified areas which were higher than those currently listed in the Catawba Equipment Qualification Criteria Manual (EQCM). Locations on Elevation 560[ft] were existing EQ HARSH zones due to the overall Total Integrated Dose (TID) with identified EQ equipment. Locations on Elevations 577[ft] and 594[ft] were existing EQ MILD zones that were projected to transition to EQ HARSH dose levels due to this issue. The operability evaluation for all identified locations resulted in a determination of Operable But Degraded/Non-Conforming for the equipment with respect to the Duke Energy EQ Program in CR C-15-00304. Followup corrective actions have been generated in CR C-15-00304 and will be tracked under the Corrective Action Program.
>
> Based on the revised dose calculations and equipment evaluations, Catawba is pursuing the re-establishment of the appropriate lead shielding in the identified Annulus penetrations to restore the dose levels in the locations to values consistent with those currently listed in the Catawba EQCM. These corrective actions will resolve LAR Commitments 11 and 12 and the equipment qualification

> items associated with EEEB-RAI 6 and EEEB-RAI 7. The work to correct the shielding of the Annulus penetrations on Unit 1 will be completed in the Unit 1 Fall 2015 Refueling Outage. With respect to the MUR Power Uprate project, Unit 1 is the only unit being uprated. The Unit 2 Annulus penetrations will also be corrected, but the Unit 2 locations do not have any impact on the MUR Power Uprate project.

The licensee further addressed this issue in Reference 6, stating, in part, that:

> Duke Energy had originally communicated to the NRC that Catawba would remediate the affected Unit 1 Annulus penetrations in the Fall 2015 Refueling Outage. However, due to outage scheduling and resource limitations, Duke Energy has elected to revise its plans and schedule relative to this effort.

Due to the above stated issue, the licensee provided the following additional regulatory commitment:

> The affected Unit 1 Annulus penetrations will be remediated by filling the penetrations with lead wool and/or brick, or other shielding material as determined appropriate, prior to implementation of the MUR power uprate.

As stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitment an implementation requirement of the MUR power uprate. The above stated commitment replaces regulatory commitments 11 and 12. The licensee justified this revision in Reference 6, stating, in part, that:

> When the identified penetrations in the Unit I reactor building are filled with lead wool and/or brick, or other shielding material as determined appropriate, to provide shielding equivalent to that of the reactor building concrete wall, the areas around these penetrations on the 577[ft] and 594[ft] elevations will be restored to an EQ Mild radiation environment. After the Unit 1 Annulus penetrations are appropriately filled, the calculated Total Integrated Doses (TIDs) for these areas will be within the current Auxiliary Building TID values in Tables 5.0-2 and 5.0-3 of the Catawba EQCM for their associated Radiation Zones (990 Rads). Because the penetration dose evaluation considered an uprated reactor power, the restoration of these areas to EQ Mild would apply for both pre- and post-MUR conditions.

In Reference 5, the licensee stated that the LEFM transducers are located in the Turbine Building and that the Turbine Building is a mild environment with respect to radiation. Per 10 CFR 50.49, mild environment equipment is excluded from the requirements for EQ. Therefore, the LEFM transducers are not applicable to the Duke Energy EQ Program due to being located in a mild environment for radiation.

Based on its review of the LAR, the licensee's response to the information requests, and subject to the licensee completing the above-stated commitment, the NRC staff has determined that the current EQ parameters remain bounding for the MUR power uprate. Further, the NRC staff has determined that the licensee has adequately evaluated the impact of the MUR power uprate on

the EQ components. Therefore, the NRC staff concludes that implementation of the MUR power uprate will have no adverse impact on the Catawba 1 EQ Program or its ability to continue to meet the requirements of 10 CFR 50.49 as implemented by the guidance of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" (Reference 52).

3.2.3.3 NRC Staff Conclusion Regarding Electrical Engineering

The NRC staff reviewed the licensee's technical evaluations described above and, based on that information, the NRC staff has determined that Catawba 1 will continue to meet the requirements of 10 CFR 50.49, 10 CFR 50.63, and GDC 17. Therefore, the NRC staff concludes that the LAR is acceptable with respect to electrical engineering evaluations.

3.2.4 Chemical Engineering and Steam Generator Integrity

The NRC staff reviewed the LAR in accordance with RIS 2002-03, concerning the following areas: (1) Chemical and Volume Control System, (2) Steam Generator Blowdown System, (3) Steam Generator Tubes, (4) Flow-Accelerated Corrosion (FAC), and (5) Protective Coating Systems (Paints) – Organic Materials.

3.2.4.1 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.

3.2.4.1.1 Regulatory Evaluation

The NRC staff has reviewed the safety-related functional performance characteristics of CVCS components. The NRC's review criteria are based on 10 CFR 50 Appendix A 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)."

3.2.4.1.2 Technical Evaluation

In its LAR, the licensee stated that accidents, transients and other UFSAR analyses were reviewed to determine the impact of the proposed MUR power uprate on the CVCS. The licensee reported that the hot- and cold-leg temperatures of the RCS is predicted to increase and decrease

by 0.5 F to 614.9 F and 555.3 F, respectively, at maximum analytical thermal power of 102 percent RTP (3479 MWt). The LAR is requesting to increase power to 3469 MWt. The RCS pressure and average temperature are indicated to stay the same at 2250 psia and 585.1 F, respectively. The licensee evaluated the effects of the MUR power uprate 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 MUR power uprate.

The NRC staff has reviewed the licensee's evaluation and confirmed that the MUR power uprate conditions will continue to be bounded by the current licensing basis for the CVCS. The licensee has demonstrated that the CVCS will continue to maintain reactor coolant system inventory and water chemistry. The NRC staff has determined that the CVCS will continue to meet system design requirements and that no new design transients will be created at MUR power uprate conditions.

3.2.4.1.3 NRC Staff Conclusion Regarding the CVCS

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed LAR on the CVCS and has determined that the licensee has adequately addressed changes to the reactor coolant and its effects on the CVCS. The NRC staff has further determined 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 MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the CVCS.

3.2.4.2 Steam Generator Blowdown System

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.

3.2.4.2.1 Regulatory Evaluation

The NRC staff's review criteria for the SGBS are based on 10 CFR 50 Appendix A, 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.

3.2.4.2.2 Technical Evaluation

In Section IV.1.A.v of Enclosure 2 to the LAR, the licensee stated, in part, that:

> The Steam Generator Blowdown System will remain within its design basis at MUR power uprate conditions....

> The Blowdown System operates continuously with the system flowrate 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 2.3 [percent] resulting from the MUR. Therefore, the Blowdown System was evaluated conservatively with a bounding increase in flow of 2.3 [percent]. The evaluated 2.3 [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.

The licensee further stated that the components of the system susceptible to flow accelerated corrosion (FAC) will continue to be managed in accordance with the FAC Program.

The NRC staff reviewed the Catawba UFSAR and confirmed that MUR power uprate conditions will continue to be bounded by the current licensing basis for the SGBS.

3.2.4.2.3 NRC Staff Conclusion Regarding the SGBS

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

3.2.4.3 Steam Generator Tubes

SG tubes constitute a large part of the RCPB. As a result, their integrity is important to the safe operation of a reactor.

3.2.4.3.1 Regulatory Evaluation

The NRC staff's review in this area covered the effects of changes in operating conditions resulting from the proposed MUR power uprate on SG materials and the SG program. The NRC staff's review criteria for the SG Program are based on the Catawba TSs. Specific review criteria for this topic are contained in the SRP, Section 5.4.2.1, "Steam Generator Materials," for the SG materials, and Section 5.4.2.2, "Steam Generator Program," for the SG program.

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

> ... to ensure that (1) the materials used to fabricate the steam generator are selected, processed, tested, and inspected to appropriate specifications, (2) the fracture toughness of the ferritic materials is adequate, (3) the design of the steam generator limits the susceptibility of the materials to degradation and corrosion, (4) the materials used in the steam generator are compatible with the environment to which they will be exposed,

> (5) the design of the secondary side of the steam generator permits the chemical or mechanical removal of chemical impurities, and (6) any degradation to which the materials are susceptible (including fracture) is avoided, can be managed through the inservice inspection program, or can be controlled through limits placed on operating parameters. Performing periodic steam generator inspections will ensure that the integrity of the steam generator is maintained at a level comparable to that in the original design requirements.

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

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

3.2.4.3.2 Technical Evaluation

Catawba 1 has 4 BWI Model CFR-80 SGs. Each SG contains 6,633 thermally-treated Alloy 690 tubes. In Section IV.1.A.vi of Enclosure 2 to its LAR the licensee provided an evaluation of the SG tubes, stating, in part, that:

> As shown in Table IV-1, the steam generator outlet pressure decreases from 1021 psia at current full power conditions to 1020.7 psia at 102 [percent] of 3411 MWt (3479 MWt) and the RCS pressure remains unchanged at 2250 psia. Therefore, the normal operating differential pressure across a steam generator tube increases from 1229 psid at current conditions to 1229.3 psid at 102 [percent] of 3411 MWt (3479 MWt).
>
> As shown in LAR Table IV-1, the feedwater temperature increases from 440 [F] at current conditions to 442 [F] at 102 [percent] of 3411 MWt (3479 MWt). As shown in this same table, the steam flow rate increases from 15.1 E6 lbm/hr at current conditions to 15.5 E6 lbm/hr at 102 [percent] of 3411 MWt (3479 MWt). The feedwater flow rate increases from 15.1 E6 lbm/hr at current conditions to 15.5 E6 lbm/hr at 102 [percent] of 3411 MWt (3479 MWt).
>
> 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.
>
> In addition, a review of calculations performed which assessed the integrity of tubes containing flaws of various types when subjected to operating and accident

loads was conducted. This review ensured that existing structural margins are maintained for the MUR power uprate design conditions.

In Reference 5, the licensee stated that an analysis had been performed to assess the potential impact of the proposed MUR power uprate on SG tube vibration response and consequential wear degradation (wear at tube support locations, tube-to-tube wear) and fatigue. Three FIV mechanisms were investigated for the tubes in the Catawba 1 SGs, namely: Fluid Elastic Instability (FEI), Random Turbulence Excitation (RTE), and Vortex Shedding. In addition, the licensee performed fatigue evaluation of the tubes for uprated conditions using the primary and secondary stresses from SG tubes. The licensee indicated that the results of the FIV analysis met the acceptance criteria, concluding that FEI and excessive RTE leading to detrimental tube wear are not predicted. Furthermore, the licensee determined that the resulting fatigue on the SG tubes, due to the power uprate, is within the allowable limits.

The NRC staff evaluated the material provided by the licensee and determined that the changes in operating conditions at LAR conditions would be small. Further, 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 has determined that the materials used in the SG remain acceptable, the fracture toughness of the ferritic materials is adequate, the design still limits the susceptibility of the materials to degradation and corrosion, the materials used in the SG remain compatible with the environment, the design permits the removal of impurities, and that any degradation that could occur is either avoided or can be managed. In addition, the NRC staff has determined that the impact of the power uprate on SG tube vibration and fatigue remains within acceptable limits for safe operation.

With respect to the SG program, the NRC staff has determined that the changes in operating conditions have no effect on the ability to implement the SG program. As a result, the NRC staff has determined that the design of the SG remains adequate for implementing the SG program. The changes in operating conditions may result in increased susceptibility to degradation and may result in increased degradation growth rates. Although this may occur, the NRC staff has determined that the SG program is still acceptable 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 MUR power uprate conditions. Although this analysis was not reviewed by the NRC staff in detail, the NRC staff has determined that the tube repair criteria remain valid under the MUR power uprate conditions. This determination is based on NRC staff's approval of 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 has determined that the SG program remains acceptable for MUR power uprate conditions.

3.2.4.3.3 NRC Staff Conclusion Regarding the SG Tubes

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

3.2.4.4 Flow-Accelerated Corrosion

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 immune 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.

3.2.4.4.1 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.

3.2.4.4.2 Technical Evaluation

In its LAR, the licensee stated that the FAC program is based on the guidelines in the Electric Power Research Institute (EPRI) Report NSAC-202L, "Recommendation for an Effective Flow-accelerated Corrosion Program" (Reference 53). The licensee stated that the proposed MUR power uprate will impact the FAC related piping wear rates; however, the changes will be small. Further, the proposed MUR power uprate will not have a significant impact on the FAC Program. The licensee stated that the impact on the future piping wear rates will be determined through the use of the CHECWORKS Steam/FW Application monitoring software. In Table IV.1.E-1 of its LAR the licensee provided an increase in current wear rate due to the MUR power uprate, showing that the largest expected wear rate increase is 6.88 percent. This relatively small increase in wear rate is within the current program's predictive capabilities. Finally, the licensee stated that the existing FAC program will incorporate all parameter changes associated with the proposed uprate conditions.

The NRC staff has determined that the current FAC program incorporates adequate conservatism to ensure that components susceptible to FAC are managed appropriately prior to exceeding minimum wall thickness. The NRC staff verified that the licensee evaluated the proposed MUR power uprate parameters using the CHECWORKS monitoring software. The NRC staff has

determined that the FAC program, with the incorporated system changes resulting from the MUR power uprate, will provide reasonable assurance that components susceptible to FAC will be managed appropriately following the implementation.

3.2.4.4.3 NRC Staff Conclusion Regarding FAC

The NRC staff has reviewed the licensee's evaluation of the proposed MUR power uprate on the FAC analysis and determined that the licensee has adequately addressed the impact of changes in plant operating conditions on the FAC analysis. The NRC staff has determined that the licensee has demonstrated the 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 MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to FAC.

3.2.4.5 Protective Coating Systems (Paints) - Organic Materials

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.

3.2.4.5.1 Regulatory Evaluation

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 on the quality assurance requirements of 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components. The NRC staff also used RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Rev. 1 (Reference 54), 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.

3.2.4.5.2 Technical Evaluation

The licensee evaluated the impact of the MUR power uprate on its containment coatings program in Section VII.6.B of Enclosure 2 to the LAR, stating, in part, that:

> The proposed MUR power uprate at [Catawba] 1 does not have any impact to the programmatic aspects of the Coatings Program. The LOCA containment response analyses remain bounding for the MUR power uprate. There were no changes to the containment analyses that would require a change to the containment design pressure or temperature. Since the containment design pressure and temperature limits were used to qualify the Service Level 1 containment coatings and those limits are not changing, the Service Level 1 containment coatings remain qualified under MUR power uprate conditions. Therefore, the MUR power uprate is bounded by current analysis of record and no changes are required.

The NRC staff has reviewed the licensee's evaluation and the Catawba 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 implementation of the proposed MUR power uprate and that temperature, pressure, and radiation limits under MUR power uprate conditions will continue to be bounded by the conditions to which the coatings were qualified.

3.2.4.5.3 NRC Staff Conclusion Regarding Protective Coating Systems

The NRC staff has reviewed the licensee's evaluation of the effects of the MUR power uprate on protective coating systems and determined 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 has further determined that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the MUR power uprate. 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 concludes that the LAR is acceptable with respect to protective coatings systems.

3.2.5 Effect of MUR Power Uprate on Major Components

3.2.5.1 Safety-Related Valves

The NRC staff reviewed the effects of the implementation of the proposed MUR power uprate on the licensee's safety-related valves analyses.

3.2.5.1.1 Regulatory Evaluation

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 NRC Generic Letter (GL) 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance" (Reference 55); GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (Reference 56); and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Reference 57).

3.2.5.1.2 Technical Evaluation

In Section IV.1.A.ix of Enclosure 2 of its LAR, the licensee stated, in part, that:

> The pressurizer code safety valves, power operated valves, and block valves located on top of the pressurizer provide over pressure protection for the RCS. The changes due to the MUR power increase that could potentially impact the pressurizer valves are RCS mass and reactor power (including RCP heat). The RCS mass does not significantly change due to the MUR power increase, based on the small changes in T_{hot} and T_{cold}. The MUR power uprate is bounded by the current design basis event transient analyses (Section II), and thus there is no adverse impact on the pressurizer overpressure protection valves from the MUR power uprate. Based on this review, it was determined that the analysis of record

for the pressurizer overpressure protection valves remains bounding at MUR power uprate conditions.

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 proposed MUR power uprate.

The licensee also evaluated the impact of the proposed MUR power uprate on the current air operated valve (AOV) program and GL 89-10 and GL 95-07 motor-operated valve (MOV) program in Section VII.6.D of its LAR. The overall system evaluations concluded that valve function, valve design, operational conditions, thrust, and torque requirements are unaffected by the MUR power uprate and all valves remain capable of performing their design basis functions. Therefore, no changes are required to the existing AOV or MOV programs.

3.2.5.1.3 NRC Staff Conclusion Regarding Safety-Related Valves

Based on its review of the licensee's evaluations, the NRC staff has determined that the performance of the safety-related valves will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the safety-related valve programs.

3.2.5.2 Reactor Coolant Pumps

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

3.2.5.2.1 Regulatory Evaluation

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

3.2.5.2.2 Technical Evaluation

In Section IV.1.A.vii of Enclosure 2 of its LAR, the licensee evaluated the impact of the proposed MUR power uprate conditions on the existing design basis analyses for the RCPs. The evaluation showed that there are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect RCP performance. The current plant design is considered bounding and requires no modifications to the RCPs.

3.2.5.2.3 NRC Staff Conclusion Regarding the RCP

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

3.2.5.3 Inservice Inspection Program

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

3.2.5.3.1 Regulatory Evaluation

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

3.2.5.3.2 Technical Evaluation

In Section IV.1.E.i of Enclosure 2 to the LAR, the licensee described its evaluation of the impact of the MUR power uprate on the ISI program for ASME Class 1, 2, and 3 components at Catawba 1, stating, in part, that:

> The ISI Program is discussed in UFSAR Section 5.2.4. ASME Class 1, 2 and 3 components are examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI in effect as specified in 10 CFR 50.55a(g) to the extent practical. The MUR power uprate conditions were reviewed for impacts on the ISI Program. The ISI Program will continue to assess the operational qualification of ASME Class 1, 2, and 3 systems. The Program does not require revision as a result of the MUR power uprate.

3.2.5.3.3 NRC Staff Conclusion Regarding the ISI Program

Based on its review of the licensee's evaluations, the NRC staff has determined that the ISI program will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the ISI Program.

3.2.5.4 Inservice Testing Program

The NRC staff reviewed the effects of the implementation of the proposed MUR power uprate on the licensee's inservice testing (IST) program.

3.2.5.4.1 Regulatory Evaluation

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

3.2.5.4.2 Technical Evaluation

In Section IV.1.E.ii 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 Catawba 1, stating, in part, that:

The IST Program establishes performance requirements for pump and valve testing. The program is addressed in Catawba Technical Specification 5.5.8. Catawba Nuclear Station has developed and implemented an IST Program for pumps and valves per these requirements. The proposed MUR power uprate does not have any impact to the programmatic aspects of the IST Program. It does not change any of the regulatory requirements of the program or in any way change the scope of the program. It does not add or delete any systems or components, since the new LEFM will not be part of the IST Program.

3.2.5.4.3 NRC Staff Conclusion Regarding the IST Program

Based on its review of the licensee's evaluations, the NRC staff has determined that the IST program will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the IST Program.

3.2.6 Neutron Fluence Evaluation

In its LAR, the licensee stated that RAPTOR-M3G was used for the neutron fluence calculations for Catawba 1 under MUR power uprate conditions. RAPTOR-M3G has not been NRC-approved as a methodology for neutron fluence calculations.

3.2.6.1 Regulatory Evaluation

The NRC staff evaluated the RAPTOR-M3G neutron fluence method in accordance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 58).

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

- Performed using an acceptable methodology
- Contains an analytic uncertainty analysis identifying possible sources of uncertainty
- Contains a benchmark comparison to approved results of a test facility
- Demonstrates plant-specific qualification by comparison to measured fluence values

3.2.6.2 Technical Evaluation

The licensee stated in its LAR that the fluence calculations were performed in accordance with the following Westinghouse Licensing Topical Reports:

- WCAP-14040-A, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Rev. 4, May 2004 (Reference 59),
- WCAP-16083-NP-A, Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry, Rev. 0, May 2006 (Reference 60), and
- WCAP- 16083-NP, Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry, Rev. 1, April 2013 (Reference 61).

The evaluation uses the three-dimensional neutron transport code RAPTOR-M3G from WCAP-16083-NP, Rev. 1, rather than the TORT code, which is a part of the approved Westinghouse methodology.

In its LAR, the licensee stated that the neutron calculations were performed in a manner consistent with the guidance set forth in RG 1.190. A solution to the Boltzmann transport equation is approximated using the three-dimensional discrete ordinates radiation transport code RAPTOR-M3G. The licensee uses a cross-section library based on the ENDF/B-VI nuclear data. Numeric approximations include a P_3 Legrende expansion for anisotropic scattering and the modeling uses S_8 order of angular quadrature. However, RG 1.190 also states that when off-midplane locations are analyzed, the adequacy of the S_8 quadrature must be demonstrated with higher-order S_n calculations. In Reference 5, the licensee demonstrated that the analysis of the Catawba vessel at off-midplane locations, shows a negligible difference between the fast fluence using the standard S_8 order of angular quadrature and a more rigorous S_n quadrature. This cross-section data and approximations were performed in accordance with the modeling guidance contained in RG 1.190. Since the licensee used RG 1.190 adherent methods to determine the vessel fluence, the NRC staff determined that the fluence calculations are acceptable.

In its LAR, the licensee further stated that space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Three dimensional flux solutions are constructed using a synthesis of azimuthal, axial, and radial flux. Source distributions include a cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions, which are used to develop spatial and energy dependent core source distributions that are averaged over each fuel cycle. This method accounts for source energy spectral effects by using an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history for each fuel assembly. The neutron source and transport calculations, as described above, were performed in accordance with the guidance set forth in RG 1.190. Based on the consistency with the guidance contained in RG 1.190, the NRC staff determined that the source and transport calculations are acceptable.

The RAPTOR-M3G method is supported by an analytic uncertainty analysis, and the estimated uncertainty is less than 20 percent, which is in accordance with RG 1.190, and therefore, acceptable. Details of the analytic uncertainty analysis are provided in WCAP-16083-NP, Rev. 1.

WCAP-16083-NP, Rev. 1 describes the methods qualification using the standard benchmark problems found in RG 1.190. The calculations were compared with the benchmark measurements from the Poolside Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL) and the H.B. Robinson Steam Electric Plant benchmark experiment. The NRC staff determined that these constitute acceptable test facilities, as they are specifically referenced in RG 1.190.

WCAP-16083-NP, Rev. 1, and WCAP-17669-NP, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," Rev. 0 (Reference 62, submitted with Reference 3), contain acceptable plant-specific benchmarking for Catawba 1 as it contains a database of PWR dosimetry benchmarking. The Catawba 1 specific geometry, a Westinghouse Reactor Pressure Vessel, is well represented

within the database. Westinghouse-specific benchmarking documented in WCAP-16083-NP-A, Rev. 0, WCAP- 16083-NP, Rev. 1, and WCAP-17669-NP, Rev. 0, indicates that surveillance capsule fluence can be calculated within 20 percent of measured values, which is in accordance with RG 1.190. Therefore, the NRC staff has determined that these uncertainties are acceptable.

3.2.6.3 NRC Staff Conclusion Regarding Neutron Fluence

As stated above, the NRC staff has determined that the neutron fluence calculation provided by the licensee in support of the MUR power uprate, adequately addresses the four criteria of RG 1.190. Therefore, the NRC staff concludes that the LAR is acceptable, with respect to the use of the RAPTOR-M3G neutron fluence calculation.

3.3 Safety Programs

3.3.1 Radiological Dose Assessment

This NRC staff review of the impact of the proposed MUR power uprate on analyzed DBA radiological consequences.

3.3.1.1 Regulatory Evaluation

The NRC staff's review of the licensee's analysis of radiological dose consequences follows the guidance of RIS 2002-03, 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.

By letter dated September 30, 2005 (Reference 63), the NRC issued license amendments to Catawba 1 and 2, which include a full-scope implementation of an AST for evaluating the consequences of DBAs. The regulatory requirements for which the NRC staff based its acceptance of the analyses are the accident dose criteria in 10 CFR 50.67, "Accident Source Term," as supplemented in Regulatory Position 4.4 of RG 1.183, and 10 CFR Part 50, Appendix A, GDC-19, "Control Room," as supplemented by SRP Section 6.4. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," in performing this review. The NRC staff also considered relevant information in the Catawba UFSAR and TSs.

The NRC staff conducted this evaluation to verify that the results of the licensee's DBA dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 and RG 1.183 after implementation of the proposed MUR power uprate.

3.3.1.2 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:

- UFSAR Section 15.1.5, "Main Steam Line Break"
- UFSAR Section 15.3.3, "Locked Rotor Accident"
- UFSAR Section 15.4.8, "Rod Ejection Accident"
- UFSAR Section 15.6.2, "Instrument Line Break"
- UFSAR Section 15.6.3, "Steam Generator Tube Rupture"
- UFSAR Section 15.6.5, "Loss of Coolant Accident"
- UFSAR Section 15.7.1, "Waste Gas Decay Tank Rupture"
- UFSAR Section 15.7.2, "Liquid Storage Tank Rupture"
- UFSAR Section 15.7.4, "Fuel Handling Accident"
- UFSAR Section 15.7.4, "Weir Gate Drop"

In its LAR, the licensee stated that the Catawba dose analysis does not change because the Catawba AOR for the UFSAR Chapter 15 analyses and other supporting analyses support the MUR power uprate (except for the LOCA peak clad temperature analysis, which is unrelated to accident dose). The licensee verified that the DBA events with radiological consequences have been conservatively performed at a power level exceeding the proposed MUR uprated power level plus uncertainty. The licensee stated that relevant DBAs, as provided in the listing above, were analyzed at 101.7 percent RTP (3469 MWt), 102 percent RTP (3479 MWt), or are not dependent on core thermal power. The MSLB and the instrument line break offsite dose analysis was performed with a reactor coolant source term that is based on the maximum reactor coolant activity allowed by Technical Specification 3.4.16. This source term is determined independent of RTP. Since there is no relation to RTP and the limit on reactor coolant activity is unchanged for the MUR uprate condition, the dose analysis remains unaffected by implementation of the MUR power uprate. The evaluation/analysis was based on the methodology, assumptions, and analytical techniques described in the applicable RGs, the SRP (where applicable), and in previous SEs.

The NRC staff confirmed that the current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 3469 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the CheckPlus LEFM system credited by the licensee. The NRC staff also confirmed that the licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM system experience operational limitations. In its LAR, the licensee indicated that a Selected Licensee Commitment will be added to address functional requirements for the LEFMs and appropriate required actions and completion times when an LEFM is non-functional. As discussed in SE Section 3.1.1.2.2.5 above, if a non-functional LEFM is not restored to functional status within 72 hours, then within 6 hours, core thermal power will be reduced to no more than 3411 MWt (the previously licensed rated thermal power).

Using the licensing basis documentation as contained in the current Catawba UFSAR, in addition to information in the LAR, the NRC staff verified that the existing Catawba UFSAR Chapter 15 radiological analyses and release assumptions bound the conditions for the proposed MUR power uprate, considering the higher accuracy of the proposed FW flow measurement instrumentation.

3.3.1.3 NRC Staff Conclusion Regarding Radiological Dose Assessment

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA dose consequence analyses at the proposed uprated power level. The NRC staff has determined that operating Catawba 1 at the proposed MUR power uprate power level will continue to meet the applicable dose limits following implementation of the MUR power uprate. The NRC staff has further determined that there is reasonable assurance that Catawba 1, following implementation of the MUR power uprate, 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

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 implementation of the proposed MUR power uprate on the plant's safe shutdown analysis to ensure that the SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire.

3.3.2.1 Regulatory Evaluation

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

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, states, in part, 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, states, in part, 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 MUR power uprate LARs identify current accident and transient AORs, which bound plant operation at the proposed uprated power level. For any DBA for which the existing AORs do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.3.2.2 Technical Evaluation

The licensee developed its LAR consistent with the guidelines in RIS 2002-03. In its LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR 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 II.1.D, item 46, and Section VII.6.A. The NRC staff also reviewed the licensee's commitment to 10 CFR 50.48 (i.e., approved fire protection program). The review covered the impact of the MUR power uprate 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 MUR power uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

In Section II.1.D, item 46 of its LAR, the licensee stated, in part, that:

> Installation of the LEFM components was reviewed under the administrative controls of the Catawba Nuclear Station design change process and found to not adversely impact safe shutdown. There are no changes to the fire detection or protection systems that could affect their safe shutdown capability. Evaluation of the fire protection systems concluded that they are not adversely affected by the MUR power uprate and are bounded by the existing design basis and analyses. Current calculations for decay heat and condensate consumption were performed at 102 [percent] of 3411 MWt (3479 MWt). Therefore, the 72-hour requirements in 10 CFR 50, App. R, Sections G.1.b and II.L are not impacted.
>
> The [Catawba] Fire Protection System is utilized for certain non-fire-protection purposes. During a B.5.b event, all AC power is lost and portable pumps are used to charge the underground fire protection system header. Catawba uses the underground fire protection system header to distribute water to meet B.5.b strategies including makeup to the spent fuel pool, the refueling water storage tank, and steam generators as well as fire suppression and Containment flooding. Operations emergency procedures for loss of feedwater provide an option to use water from the fire protection system header to make up to the steam generators.

The NRC staff has reviewed the information provided in the LAR and verified that it involves no changes to the fire protection program that may adversely impact the post-fire safe shutdown capability in accordance with 10 CFR 50, Appendix R. The NRC staff has further verified that the proposed MUR power uprate is bounded by the existing design basis and analyses and, therefore, does not adversely impact the post-fire safe shutdown capability.

During its review, the NRC staff noted that the licensee did not discuss the impact of other uses of fire water for non-fire protection purposes at Catawba, or how it may impact the need to meet the fire protection system design demands. Due to this, the NRC staff requested that the licensee provide additional information, discussing how any non-fire suppression use of fire protection water will impact the ability to meet the fire protection system design demands.

In Reference 3, the licensee responded to this information request, stating, in part, that:

> An evaluation of fire protection systems concluded that they were not adversely affected by the MUR power uprate since supporting calculations were performed at 102 [percent] of 3411 MWt (3479 MWt).
>
> As discussed in the Catawba MUR LAR, Enclosure 2, Section 46, Safe Shutdown Fire, the Catawba Fire Protection System is utilized for certain non-fire protection uses as well as for fire suppression. These non-fire protection uses of fire water were reviewed and found not to be impacted by the MUR power uprate.

The NRC staff has determined that the response to the request is acceptable based on its conclusion that non-fire protection aspects of the fire protection system for activities other than fire protection activities, would not impact the fire protection system design demands under the proposed MUR power uprate condition.

Further, in Section VII.6.A of Enclosure 2 of its LAR, the licensee stated, in part, that:

> The MUR power uprate does not change or modify the credited equipment necessary for post fire safe shutdown nor does it reroute essential cables or relocate essential components credited by the safe shutdown analysis. Installation of the LEFM components was reviewed under the administrative controls of the Catawba Nuclear Station design change process and found to not adversely impact safe shutdown. Additional building heat-up will be minimal such that currently credited fire protection manual actions will not be prevented from being accomplished by their required time. Damage control procedures have actions to open doors, bring in fans, or use other methods to cool the environment for more suitable working conditions and to ensure proper operation of safe shutdown equipment. No new operator actions were identified.

Based on the licensee's fire-related safe-shutdown assessment, the NRC staff has determined that the licensee has adequately accounted for the effects of the implementation of the MUR power uprate on the ability of the required fire protection systems to achieve and maintain safe shutdown conditions. The NRC staff has also determined that this aspect of the capability of the associated SSCs to perform their design-basis functions after implementation of the MUR power uprate is acceptable with respect to fire protection. Further, the NRC staff has reviewed the information provided by the licensee concerning the fire protection program elements listed above, and has verified that they are not impacted by the MUR power uprate. Note that this SE does not approve any new or existing operator manual actions concerning the Catawba 1 and 2 fire safe-shutdown analysis.

3.3.2.3 NRC Staff Conclusion Regarding Fire Protection

Based on its review, the NRC staff has determined that implementation of the MUR power uprate will not have a significant impact on the fire protection program or post-fire safe shutdown capability at Catawba 1. Therefore, the NRC staff concludes that the LAR is acceptable with respect to fire protection.

3.3.3 Human Factors

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

3.3.3.1 Regulatory Evaluation

Guidance for reviewing the licensee's human factors evaluation associated with the implementation of an MUR power uprate is available in RIS 2002-03.

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 power uprates in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4. The licensee's evaluation of human factors issues concerning implementation of the MUR power uprate are described in Sections VII.1 through VII.4 of Enclosure 2 to the LAR. The following sections evaluate the licensee's analysis of these human factors issues.

3.3.3.2.1 Operator Actions

Question: Has the licensee made a statement confirming that operator actions that are sensitive to the power uprate, including any effects on the time available for operator actions, have been identified and evaluated?

Response: Yes, in Section VII.1 of Enclosure 2 of its LAR, the licensee states, in part, that:

> The proposed MUR power uprate will be implemented under the administrative controls of the Catawba Nuclear Station design change process. The design change process ensures any impacted normal, abnormal and emergency operating procedures having operator actions are revised prior to the implementation of the MUR power uprate if required. An evaluation was performed of the Operator Actions and no impacts were identified.
>
> Time Critical Operator Actions (TCOA) are associated with the mitigation of postulated events.

> These actions must be performed in a specified time in order to assure the plant complies with assumptions made during the analysis of these postulated events. The TCOA were evaluated individually in system evaluations and against the Catawba licensing analyses presented in Section II of this enclosure to ensure they remain bounded. All of the TCOAs remain unchanged following the MUR power uprate.

The NRC staff has reviewed the licensee's statements relating to any impacts of the MUR power uprate on operator actions. The NRC staff has determined that the proposed MUR power uprate will not adversely impact operator actions or their response times because there are no changes required. Therefore, the NRC staff concludes that the statements provided by the licensee conform to Section VII.1 of Attachment 1 to RIS 2002-03.

3.3.3.2.2 Emergency and Abnormal Operating Procedures

Question: Has the licensee made a statement confirming that it has identified all required changes to the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) to ensure that changes to the EOPs and/or AOPs do not adversely affect defense in depth or safety margins?

Response: Yes, in Section VII.2.A of Enclosure 2 of its LAR, the licensee states, in part, that:

> The proposed MUR power uprate will be implemented under the administrative controls of the Catawba Nuclear Station design change process. EDM 601, Engineering Change Manual, provides the administrative controls relevant to identifying impacted procedures, controls, displays, alarms, the Operator Aid Computer (which includes the Safety Parameter Display System), and other operator interfaces, the simulator, and training. The design change process ensures any impacted emergency and abnormal operating procedures are revised prior to the implementation of the power uprate.

The NRC staff has reviewed the information provided by the licensee and concludes that the statements provided by the licensee are in conformance with Section VII.2.A of Attachment 1 to RIS 2002-03.

3.3.3.2.3 Changes to Control Room Controls, Displays, and Alarms

Question: Has the licensee made a statement confirming that it has identified all required changes to the control room controls, displays (including the safety parameter display system), and alarms to ensure that any required changes do not adversely affect defense in depth or safety margins?

Response: Yes, in Section VII.2.B of Enclosure 2 of its LAR, the licensee states, in part, that:

> A review of plant systems has indicated that only minor modifications are necessary (e.g., software modification that redefines the new 100 [percent] RTP).

> Catawba Nuclear Station follows the established engineering procedures (EDM 601, Engineering Change Manual, as noted in Section VII.2.A) to ensure the necessary minor modifications are installed prior to implementing the proposed power uprate.
>
> An "LEFM System Trouble" alarm window will be added to the control room alarm panel to alert the operator when there is a problem with the LEFM.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the control room. The NRC staff has determined that the proposed changes are minimal and do not present any adverse effects to the operators' functions in the control room. The NRC staff concludes that the information provided by licensee is in conformance with Section VII.2.B of Attachment 1 to RIS 2002-03.

3.3.3.2.4 Control Room Plant Reference Simulator

Question: Has the licensee made a statement confirming that it has identified all required changes to the control room plant reference simulator to ensure that any required changes do not adversely affect defense in depth or safety margins?

Response: Yes, in Section VII.2.C of Enclosure 2 of its LAR, the licensee states, in part, that:

> A review of the plant simulator will be conducted, and necessary changes made, under the administrative controls (EDM 601, Engineering Change Manual, as noted in Section VII.2.A) of the Catawba Nuclear Station.

The NRC staff has reviewed the licensee's evaluation of proposed changes to the plant simulator related to the MUR power uprate. The NRC staff concludes that the statements provided by the licensee conform with Section VII.2.C of Attachment 1 to RIS 2002-03.

3.3.3.2.5 Operator Training

Question: Has the licensee made a statement confirming that it has identified all required changes to the operator training program to ensure that any required changes do not adversely affect defense in depth or safety margins?

Response: Yes, in Section VII.2.D of Enclosure 2 of its LAR, the licensee states, in part, that:

> Operator training on the plant changes required to support the MUR power uprate will be completed prior to MUR power uprate implementation.
>
> Training on operation and maintenance of the Caldon LEFM CheckPlus System, will be developed and completed prior to implementation of the MUR power uprate.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the operator training program. The NRC staff concludes that the statements provided by the licensee are in conformance with Section VII.2.D of Attachment 1 to RIS 2002-03.

3.3.3.2.6 Modifications

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

Response: Yes, in Section VII.3 of Enclosure 2 of its LAR, the licensee states, in part, that:

> All changes/modifications to the simulator and the associated manuals and instructional materials will be implemented in accordance with the Catawba engineering change process (EDM 601, Engineering Change Manual, as noted in Section VII.2.A) to capture all plant changes as a result of the MUR power uprate. Duke Energy will complete all modifications identified in Section VII.2.B related to the MUR power uprate and complete the training of operators, prior to implementation of the power uprate.

The NRC staff has reviewed the LAR and concludes that the statements provided by the licensee are in conformance with Section VII.3 of Attachment 1 to RIS 2002-03.

3.3.3.2.7 Temporary Operation above Licensed Full Power Level

Question: Has the licensee made a statement confirming its intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level? The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

Response: Yes, in Section VII.4 of Enclosure 2 of its LAR, the licensee states, in part, that:

> Operating Procedures (OPs) have been reviewed and required changes will be documented and implemented as part of the normal Engineering Change process (EDM 601, Engineering Change Manual, as noted in Section VII.2.A), in particular, the procedure related to temporary operation above full steady-state licensed power levels will be reviewed and modified as necessary.

The NRC staff has reviewed the LAR and concludes that the statements provided by the licensee are in conformance with Section VII.4 of Attachment 1 to RIS 2002-03.

3.3.3.2.8 Regulatory Commitments

Associated with the its response to RIS 2002-03, Attachment 1, Section VII, Items 1 through 4, the licensee provided the following regulatory commitments, commitments 2 and 5:

> Duke Energy will implement modifications associated with the MUR power uprate as discussed in Enclosure 2, VII.2.A (emergency and abnormal operating procedures), VII.2.B (control room controls, displays and alarms), VII.2.C (control room plant reference simulator), and VII.2.D (operator training program).
>
> An "LEFM System Trouble" alarm window will be added to the control room alarm panel to alert the operator when there is a problem with the LEFM.

As stated in Section 4.0 of this SE, the NRC staff has made completion of the above regulatory commitments an implementation requirement of the MUR power uprate.

3.3.3.3 NRC Staff Conclusion Regarding Human Factors

The NRC staff has completed its human factors review of the LAR and has determined that the licensee has adequately considered, or will consider, the impact of the MUR power uprate on operator actions, EOPs and AOPs, control room components, the plant reference simulator, and operator training programs. Therefore, the NRC staff concludes that the LAR is acceptable regarding the Human Factors review and that the statements provided by the licensee are in conformance with Section VII of Attachment 1 to RIS 2002-03.

3.4 Changes to the RFOL and Technical Specifications associated with the MUR Power Uprate

3.4.1 Regulatory Evaluation

The NRC staff's guidance for reviewing changes to the TS is contained in 10 CFR 50.36, "Technical Specifications," where the NRC established its regulatory requirements related to the content of TS. Specifically, 10 CFR 50.36(c) requires that TS include items in the following categories: safety limits, limiting safety system settings, and limiting control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

3.4.2 Technical Evaluation

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

> Operating License Page 3 – Maximum Power Level
>
> For the Catawba Unit 1 operating license, the steady state licensed power level will change from 3411 MWt to 3469 MWt.
>
> TS 1.1, Definition of Rated Thermal Power
>
> RATED THERMAL POWER will change from 3411 MWt to 3469 MWt.

<u>TS Figure 3.4.3-1, RCS Heatup Limitations, and Figure 3.4.3-2, RCS Cooldown Limitations</u>

The Unit 1 heatup and cooldown limit figures are revised to reflect the new limit of applicability of 30.7 EFPY versus 34 EFPY.

<u>TS Table 3.7.1-1, OPERABLE Main Steam Safety Valves (MSSVs) versus Maximum Allowable Power Range Neutron Flux High Setpoints in Percent of RATED THERMAL POWER</u>

As discussed in Technical Specification (TS) Bases 3.7.1, Actions A.1 and A.2, operation with one or more MSSVs inoperable is permissible if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. The basis for determining the reduced high flux trip setpoint is detailed in TS Bases 3.7.1, Actions A.1 and A.2. With the MUR uprate, there is an increase in steam flow as shown in Enclosure 2, Table IV-1. Revised maximum allowable power range neutron flux high setpoints were calculated and resulted in changes to TS Table 3.7.1-1 with 4 and 3 MSSVs per steam generator OPERABLE. A separate column for Catawba Unit 1 setpoints was added.

As discussed throughout this SE, the NRC staff has determined that the licensee's proposal to increase the RTP from 3411 MWt to 3469 MWt as part of an MUR power uprate is acceptable. Therefore, the NRC staff has determined that changing the RTP as stated in the RFOL and in TS 1.1, Definitions, RATED THERMAL POWER, from 3411 MWt to 3469 MWt, is acceptable.

As discussed in Section 3.2.1.2 of this SE, the NRC staff has determined that the changes to the Catawba 1 P-T Limits Curves to reflect the new limit of applicability of 30.7 EFPY versus 34 EFPY, is acceptable.

Concerning the proposed changes to TS Table 3.7.1-1, the NRC staff reviewed the analyses in Chapter 15 of the Catawba UFSAR. For excessive increase in secondary steam flow and the loss of normal feedwater flow, the analyses are performed at 101.7 percent RTP (3469 MWt) or at 102 percent RTP (3479 MWt). The NRC staff has determined that these analyses bound the proposed MUR power uprate power level of 3469 MWt. Additionally, the proposed percent RTP setpoints for 4 operable MSSVs and 3 operable MSSVs are reduced from 58 to 57 percent RTP and from 41 to 40 percent RTP, respectively for Catawba 1. This reduction is in the conservative direction when compared to the original percent RTP setpoints. The proposed percent RTP setpoint for 2 operable MSSVs remains the same after implementation of the MUR power uprate. This is acceptable because the additional relief capacity provided by the 2 MSSVs at the same percent RTP (but higher power level) is within the rated capacity of the MSSVs per ASME design codes (i.e., 110 percent). Thus, the proposed changes to Table 3.7.1-1 are acceptable. The percent RTP versus minimum number of MSSVs operable setpoints as shown in TS Table 3.7.1-1 do not change for Catawba 2.

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

3.4.3 Conclusion

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

3.5 NRC Staff's Technical Conclusion

The NRC staff confirmed that the licensee provided all of the information requested by RIS-2002-03 in order to justify a smaller margin for power measurement uncertainty. As the methodology used to quantify the uncertainty in the reactor thermal power uncertainty calculation is consistent with the limitations and conditions in the NRC-approved topical reports, the NRC staff has determined that the licensee may apply a reduced margin for power measurement uncertainty consistent with Appendix K of 10 CFR 50. Therefore, the NRC staff concludes that the licensee's request to correspondingly uprate the current licensed power from 3411 MWt to 3469 MWt and the associated changes to the TSs for Catawba 1 are acceptable.

4.0 REGULATORY COMMITMENTS

In Attachment 1 to its LAR, the licensee provided a list of regulatory commitments. Reference 6 provided an additional commitment that replaced two of the previously made commitments, as discussed in Section 3.2.3.3 of this SE. The complete list is provided below.

Table 4-1 – Regulatory Commitments

	Commitment	Completion Date
1	Any revisions to setpoint calculations or calibration procedures necessary to reflect the increased rated thermal power will be implemented. All maintenance procedures for the new equipment added for the MUR power uprate will be implemented.	Prior to implementation of the MUR power uprate.
2	Duke Energy will implement modifications associated with the MUR power uprate as discussed in Enclosure 2, VII.2.A (emergency and abnormal operating procedures), VII.2.B (control room controls, displays and alarms), VII.2.C (control room plant reference simulator), and VII.2.D (operator training program).	Prior to implementation of the MUR power uprate.
3	Acceptance testing following installation of the CheckPlus systems in Catawba Unit 1 will confirm that as built parameters are within the bounds of the error analyses.	Prior to implementation of the MUR power uprate.
4	A Selected Licensee Commitment will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is non-functional.	Prior to implementation of the MUR power uprate.
5	An "LEFM System Trouble" alarm window will be added to the control room alarm panel to alert the operator when there is a problem with the LEFM.	Prior to implementation of the MUR power uprate.
6	The procedure related to temporary operation above full steady-state licensed power levels will be reviewed and modified as necessary.	Prior to implementation of the MUR power uprate.

	Commitment	Completion Date
7	After the LEFM CheckPlus system is installed and operational, thirty days of data will be collected comparing the LEFM CheckPlus operating data to the venturi data to verify consistency between the thermal power calculation based on the LEFM and other plant parameters.	Prior to implementation of the MUR power uprate.
8	A modification is planned to extend the outside air intakes for the Control Room and Control Room Area Pressurizing Subsystem to ensure the distance between the source-receptor pair is separated by 10 meters. This modification will ensure the design margin is maintained in the MSLB radiological dose calculation for the control room TEDE. The MUR power uprate will not impact this modification.	This modification is required to be completed prior to implementation of the MUR power uprate.
9	Duke Energy will resolve the issue of the qualification of the six ITT Barton pressure transmitters in the Reactor Vessel Level Indication System that was identified by the EQ review for being qualified to the post-MUR power uprate TID.	Prior to implementation of the MUR power uprate.
10	Duke Energy will resolve the issue of the qualification of the Struthers Dunn Type 219 relay that was identified by the EQ review for being qualified to the post-MUR power uprate TID.	Prior to implementation of the MUR power uprate.
11	Duke Energy will resolve the issue of portions of one area (Radiation Zone 30 in the Catawba Auxiliary Building at the 577 foot elevation) that were found to exceed the normal operating 40-year dose listed in the Catawba Environmental Qualification Criteria Manual for pre-MUR power uprate conditions. –**Replaced by Commitment 15**	Prior to implementation of the MUR power uprate.
12	Duke Energy will resolve the issue of portions of one area (Radiation Zone 45 in the Catawba Auxiliary Building at the 594 foot elevation) that were found to potentially exceed the normal operating 40-year dose listed in the Catawba Environmental Qualification Criteria Manual for post-MUR power uprate conditions. –**Replaced by Commitment 15**	Prior to implementation of the MUR power uprate.
13	Duke Energy will re-evaluate the Loss-of-Coolant Accidents (UFSAR Section 15.6.5) consistent with the reload methodology.	Prior to implementation of the MUR power uprate.
14	Duke Energy will evaluate the 50 equipment IDs and the 40 enclosures containing 21 individual component types encompassing 330 total components for post-MUR EQ conditions.	Prior to implementation of the MUR power uprate.
15	The affected Unit 1 Annulus penetrations will be remediated by filling the penetrations with lead wool and/or brick, or other shielding material as determined appropriate. Replaces Commitments 11 and 12.	Prior to implementation of the MUR power uprate.

An implementation requirement is included in Section 3 of the License Amendments to ensure that the above Regulatory Commitments are completed coincident with the implementation of the MUR power uprate.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on November 4, 2014 (79 FR 65429). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

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Date: April 29, 2016

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
AAC	alternate AC
AC	alternating current
AFW	auxiliary feedwater
AOP	abnormal operating procedure
AOR	analysis of record
AOT	allowed outage time
AOV	air-operated valve
ARL	Alden Research Laboratories
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
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
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
EQ	environmental qualification
ESF	engineered safety feature
F	degrees Fahrenheit
FAC	flow accelerated corrosion
FEI	Fluid Elastic Instability
FIV	flow induced vibration
FR	flatness ratio
FW	feedwater
GDC	General Design Criteria
gpm	gallons per minute
HELB	high energy line break
HVAC	heating, ventilation and air conditioning
I&E	inspection and evaluation
ISA	Instrument Society of America
ISI	inservice inspection
IST	inservice test
kV	kiloVolt
LAR	license amendment request
LBB	leak-before-break

LEFM	leading edge flow meter
LOCA	loss-of-coolant accident
LTOPS	low temperature overpressure protection system
M&E	mass and energy
Mlbm/hr	million pounds per hour
MELB	moderate energy line breaks
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
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
NSW	nuclear service water
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
PU	per unit
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCCA	Rod Cluster Control Assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFOL	Renewed Facility Operating License
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
SE	safety evaluation
SFP	spent fuel pool
SG	Steam Generator

SGBS	Steam Generator Blowdown System
SLC	Select Licensee Commitments
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
TEDE	Total Effective Dose Equivalent
TID	Total Integrated Dose
TS	Technical Specification
TSO	Transmission System Operator
UHS	Ultimate Heat Sink
UFM	ultrasonic flow meters
UFSAR	updated final safety analysis report
UPS	uninterruptible power supply
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 regarding this matter, I may be reached at (301) 415-4090 or by e-mail at jeffrey.whited@nrc.gov.

Sincerely,

/RA/

Jeffrey A. Whited, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 281 to RFOL NPF-35
2. Amendment No. 277 to RFOL NPF-52
3. Safety Evaluation

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AObodoako, NRR
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ADAMS Accession No.: ML16081A333 *via memo dated

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/ARCB/BC*	NRR/AFPB/BC*	NRR/AHPB/BC(A)*	NRR/EEEB/BC*
NAME	JWhited	SFigueroa	UShoop	AKlein	DChung	JZimmerman
DATE	04/29/2016	03/29/2016	02/02/2015	01/26/2015	08/01/2014	01/15/2016
OFFICE	NRR/EMCB/BC*	NRR/EPNB/BC	NRR/SRXB/BC(A)	NRR/ESGB/BC*	NRR/SCVB/BC*	NRR/SBPB/BC*
NAME	TLupold	DAlley	EOesterle	GKulesa	RDennig	GCasto
DATE	01/13/2015	03/23/2016	03/23/2016	05/18/2015	02/18/2015	02/04/2015
OFFICE	NRR/STSB/BC (A)	NRR/EVIB/BC	NRR/EICB/BC	OGC (NLO)	NRR/LPL2-1/BC	DORL/DD
NAME	SAnderson (MHamm for)	JMcHale (JPoehler for)	JThorp*	JWachutka (w/comments)	MMarkley	ABoland
DATE	03/31/2016	03/23/2016	06/11/2015	04/11/2016	04/25/2016	04/28/2016

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