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

October 22, 2009

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME0965 AND ME0966)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 257 and 238 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated March 26, 2009, as supplemented by letters dated July 8, 16, and 24, 2009.

These amendments revise the licenses and TSs to reflect an increase in the rated thermal power from 2893 megawatts thermal (MWt) to 2940 MWt (1.6 percent increase). The increase is based upon increased feedwater flow measurement accuracy achieved by using high-accuracy Caldon CheckPlus$^{TM}$ Leading Edge Flow Meter ultrasonic flow measurement instrumentation.

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

Sincerely,

V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 257 to NPF-4
2. Amendment No. 238 to NPF-7
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 257
Renewed License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated March 26, 2009, as supplemented by letters dated July 8, 16, and 24, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by adding License Condition 2.H. to Facility Operating License No. NPF-4, as indicated in the attachment to this license amendment, and reads as follows:

H.

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 1. Dominion will perform the final acceptance of the North Anna 1 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1). | Prior to operating above 2893 MWt (98.4% RTP). |
| 2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0). | Prior to operating above 2893 MWt (98.4% RTP). |
| 3. Procedures and documents for the new UFM (Attachment 5 Section I.1.D.1.1, I.1.H, and VII.2.A). | Prior to operating above 2893 MWt (98.4% RTP). |
| 4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, and VII.2.D). | Prior to operating above 2893 MWt (98.4% RTP). |
| 5. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 6. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4). | Prior to operating above 2893 MWt (98.4% RTP). |
| 7. Replace Steam Generator secondary manway bolts or change cumulative fatigue usage analysis to support using existing bolts for the licensed period for each unit (Attachment 5 Section IV.1.A.vi.2 and IV.1.B.ii). | Prior to exceeding 45 years of in-service use for each secondary manway bolt. |

H. (continued)

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 8. The impact of radiation effects on the EQ Program qualification requirements will be determined (Attachment 5 Section V.1.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 9. The FAC Checkworks SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii). | Prior to operating above 2893 MWt (98.4% RTP). |
| 10. Dominion will determine the EQ-service life of the excore detectors. (Attachment 5 Section II.2). | Prior to operating above 2893 MWt (98.4% RTP). |
| 11. Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments | Prior to operating above 2893 MWt (98.4% RTP). |
| 12. Confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour completion time | Prior to any such use above 2893 MWt (98.4% RTP). |

3. The license is also amended by 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-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 257 , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license is also amended by a change to paragraph 2.C.(1) of Renewed Facility Operating License No. NPF-4 as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (1) Maximum Power Level

   VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2940 megawatts (thermal).

5. This license amendment is effective as of its date of issuance and shall be implemented by July 14, 2010. Accordingly, the scheduled completion dates listed in License Condition 2.H., shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the condition and shall determine the environmental qualification service life of the excore detectors and incorporate changes in the qualified lifetime of this equipment into environmental qualification program documentation, prior to operating above the current maximum operating level of 2893 MWt, as described in Virginia Electric and Power Company's letters dated March 26, 2009, July 8, 2009, and July 24, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G Howe for

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-4
and the Technical Specifications

Date of Issuance: October 22, 2009

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 238
Renewed License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated March 26, 2009, as supplemented by letters dated July 8, 16, and 24, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by adding License Condition 2.H. to Facility Operating License No. NPF-7, as indicated in the attachment to this license amendment, and reads as follows:

H.

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 1. Dominion will perform the final acceptance of the North Anna 2 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1). | Prior to operating above 2893 MWt (98.4% RTP). |
| 2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0). | Prior to operating above 2893 MWt (98.4% RTP). |
| 3. Procedures and documents for the new UFM (Attachment 5 Section I.1.D.1.1, I.1.H, and VII.2.A). | Prior to operating above 2893 MWt (98.4% RTP). |
| 4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, and VII.2.D). | Prior to operating above 2893 MWt (98.4% RTP). |
| 5. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 6. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4). | Prior to operating above 2893 MWt (98.4% RTP). |
| 7. Replace Steam Generator secondary manway bolts or change cumulative fatigue usage analysis to support using existing bolts for the licensed period for each unit (Attachment 5 Section IV.1.A.vi.2 and IV.1.B.ii). | Prior to exceeding 45 years of in-service use for each secondary manway bolt. |

H. (continued)

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 8. The impact of radiation effects on the EQ Program qualification requirements will be determined (Attachment 5 Section V.1.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 9. The FAC Checkworks SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii). | Prior to operating above 2893 MWt (98.4% RTP). |
| 10. Dominion will determine the EQ-service life of the excore detectors. (Attachment 5 Section II.2). | Prior to operating above 2893 MWt (98.4% RTP). |
| 11. Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments | Prior to operating above 2893 MWt (98.4% RTP). |
| 12. Confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour completion time | Prior to any such use above 2893 MWt (98.4% RTP). |

3. The license is also amended by 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-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 238 , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license is also amended by a change to paragraph 2.C.(1) of Renewed Facility Operating License No. NPF-7 as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (1) Maximum Power Level

   VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

5. This license amendment is effective as of its date of issuance and shall be implemented by July 14, 2010. Accordingly, the scheduled completion dates listed in License Condition 2.H., shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the condition and shall determine the environmental qualification service life of the excore detectors and incorporate changes in the qualified lifetime of this equipment into environmental qualification program documentation, prior to operating above the current maximum operating level of 2893 MWt, as described in Virginia Electric and Power Company's letters dated March 26, 2009, July 8, 2009, and July 24, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Howe for

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-7
and the Technical Specifications

Date of Issuance: October 22, 2009

ATTACHMENT

TO LICENSE AMENDMENT NO. 257

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 238

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Licenses and the Appendix "A" Technical Specifications (TSs) with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

| Remove Pages | Insert Pages |
|---|---|
| Licenses | Licenses |
| No. NPF-4 | No. NPF-4 |
| page 3 | page 3 |
| page 8 | page 8 |
| ----- | page 9 |
| | |
| No. NPF-7 | No. NPF-7 |
| page 3 | page 3 |
| page 8 | page 8 |
| ----- | page 9 |
| | |
| TSs | TSs |
| 1.1-5 | 1.1-5 |

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO 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;

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

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO 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 component; and

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 257, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Actions to minimize release to include consideration of:

a. Water spray scrubbing

b. Dose to onsite responders

H.

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 1. Dominion will perform the final acceptance of the North Anna 1 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1). | Prior to operating above 2893 MWt (98.4% RTP). |
| 2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0). | Prior to operating above 2893 MWt (98.4% RTP). |
| 3. Procedures and documents for the new UFM (Attachment 5 Section I.1.D.1.1, I.1.H, and VII.2.A). | Prior to operating above 2893 MWt (98.4% RTP). |
| 4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, and VII.2.D). | Prior to operating above 2893 MWt (98.4% RTP). |
| 5. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 6. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4). | Prior to operating above 2893 MWt (98.4% RTP). |
| 7. Replace Steam Generator secondary manway bolts or change cumulative fatigue usage analysis to support using existing bolts for the licensed period for each unit (Attachment 5 Section IV.1.A.vi.2 and IV.1.B.ii). | Prior to exceeding 45 years of in-service use for each secondary manway bolt. |

H. (continued)

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 8. The impact of radiation effects on the EQ Program qualification requirements will be determined (Attachment 5 Section V.1.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 9. The FAC Checkworks SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii). | Prior to operating above 2893 MWt (98.4% RTP). |
| 10. Dominion will determine the EQ-service life of the excore detectors. (Attachment 5 Section II.2). | Prior to operating above 2893 MWt (98.4% RTP). |
| 11. Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments | Prior to operating above 2893 MWt (98.4% RTP). |
| 12. Confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour completion time | Prior to any such use above 2893 MWt (98.4% RTP). |

I. This renewed license is effective as of the date of issuance and shall expire at midnight on April 1, 2038.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by:

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan

Date of Issuance: March 20, 2003

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

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

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

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as 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

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No238 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

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

a. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the

(3) Actions to minimize release to include consideration of:

a. Water spray scrubbing

b. Dose to onsite responders

H.

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 1. Dominion will perform the final acceptance of the North Anna 2 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1). | Prior to operating above 2893 MWt (98.4% RTP). |
| 2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0). | Prior to operating above 2893 MWt (98.4% RTP). |
| 3. Procedures and documents for the new UFM (Attachment 5 Section I.1.D.1.1, I.1.H, and VII.2.A). | Prior to operating above 2893 MWt (98.4% RTP). |
| 4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, and VII.2.D). | Prior to operating above 2893 MWt (98.4% RTP). |
| 5. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 6. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4). | Prior to operating above 2893 MWt (98.4% RTP). |
| 7. Replace Steam Generator secondary manway bolts or change cumulative fatigue usage analysis to support using existing bolts for the licensed period for each unit (Attachment 5 Section IV.1.A.vi.2 and IV.1.B.ii). | Prior to exceeding 45 years of in-service use for each secondary manway bolt. |

H. (continued)

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 8. The impact of radiation effects on the EQ Program qualification requirements will be determined (Attachment 5 Section V.1.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 9. The FAC Checkworks SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii). | Prior to operating above 2893 MWt (98.4% RTP). |
| 10. Dominion will determine the EQ-service life of the excore detectors. (Attachment 5 Section II.2). | Prior to operating above 2893 MWt (98.4% RTP). |
| 11. Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments | Prior to operating above 2893 MWt (98.4% RTP). |
| 12. Confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour completion time | Prior to any such use above 2893 MWt (98.4% RTP). |

I. This renewed license is effective as of the date of issuance and shall expire at midnight on August 21, 2040.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by:

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan

Date of Issuance: March 20, 2003

1.1 Definitions

| | |
|---|---|
| PHYSICS TESTS (continued) | b. Authorized under the provisions of 10 CFR 50.59; or<br>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 2940 MWt. |
| REACTOR TRIP SYSTEM (RTS) RESPONSE TIME | The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:<br>a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and<br>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. |

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR EGULATION

RELATED TO AMENDMENT NO. 257

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 238

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

## 1.0 INTRODUCTION

By letter dated March 26, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090900055), as supplemented by letters dated July 8, 2009 (ADAMS Accession No. ML091910204), July 16, 2009 (ADAMS Accession No. ML091970595), and July 24, 2009 (ADAMS Accession No. ML092050714), Virginia Electric and Power Company (the licensee) submitted a request for changes to the North Anna Power Station, Unit Nos. 1 and 2 (North Anna Units 1 and 2), Technical Specifications (TSs). The requested changes would increase the rated thermal power from 2893 megawatts thermal (MWt) to 2940 MWt, an increase of approximately 1.6 percent. The supplements dated July 8, 16, and 24, 2009, provided clarifying information that did not change the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

The proposed changes would revise the license and TSs to reflect an increase in the rated thermal power from 2893 to 2940 megawatts thermal (MWt). The electrical generator output will be revised from 965 megawatts electric (MWe) to 980.5 MWe (an increase of 15.5 MWe) for North Anna Unit 1 and from 965 MWe to 972.9 MWe (an increase of 7.9 MWe) for North Anna Unit 2. The increase is based upon increased feedwater flow measurement accuracy achieved by using high-accuracy Caldon CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement (UFM) instrumentation. This type of application is commonly referred to as a measurement uncertainty recapture (MUR) power uprate (PU). The licensee developed the application using the guidance of U.S. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

Specifically, the licensee proposes the following changes:

- Paragraphs 2.C.(1), "Maximum Power level", of the Unit 1 and Unit 2 Operating Licenses Nos. NPF-4 and NPF-7 (page 3) are revised to authorize operation of the facilities at a steady state reactor core thermal power level not in excess of 2940 MWt.
- The definition of RATED THERMAL POWER (RTP) in TS 1.1, page 1.1-5, is revised to reflect an increase from 2893 MWt to 2940 MWt.

Neutron flux instrumentation is calibrated to the core thermal power which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). This calculation is called "secondary calorimetric" for a pressurized-water reactor (PWR). The accuracy of this calculation depends primarily upon the accuracy of feedwater flow and feedwater net enthalpy measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of this parameter will result in an accurate determination of core thermal power.

This power uprate is based on a reduced measurement uncertainty of core thermal power resulting from the installation of a Caldon Leading Edge Flow Meter (LEFM) CheckPlus System to measure feedwater flow and temperature at North Anna Units 1 and 2. The licensee's submittal of Cameron (formerly Caldon) Topical Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the System," issued March 1997 (1), and its supplement, Topical Report ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √TM or LEFM CheckPlusTM System," issued October 2001 (2). These topical reports, which are generically applicable to nuclear power plants, document the ability of the Caldon LEFM Check and CheckPlus Systems to increase the accuracy of flow measurement. The NRC approved Topical Report ER-80P and its supplement, ER-157P, in safety evaluation reports (SERs) issued March 1999 (3) and December 2001 (4), respectively. Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Caldon LEFM Check System in a typical two-loop pressurized-water reactor (PWR) or two-feedwater-line boiling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. Its supplement, Topical Report ER-157P, describes the Caldon LEFM CheckPlus System and lists nonproprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using either the Caldon LEFM Check or LEFM CheckPlus System. Together, these two reports provide a generic basis for a measurement uncertainty recapture power uprate.

Cameron Engineering Reports ER-646, Revision 2, "Bounding Uncertainty Analysis for ThermalPower Determination at North Anna Unit 1 Using the LEFM CheckPlus System," and ER-637, Revision 3, "Bounding Uncertainty Analysis for Thermal Power Determination at North Anna Unit 2 Using the LEFM CheckPlus System," issued January 2009, describe the plant-specific bases for the proposed uprate at North Anna and were provided with the March 26, 2009, license amendment request. The licensee's July 8, 2009, response to the staff's request for additional information cited minor corrections to these documents and provided revised pages for ER-646, Revision 3, and ER-637, Revision 4.

The NRC has recently issued similar MUR power uprate license amendments for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 on July 22, 2009 (ADAMS Accession No. ML0918203660), Crystal River, Unit 3 on December 26, 2007 (ADAMS Accession No. ML073600419), Vogtle Electric Generating Plant, Units 1 and 2 on February 27, 2008 (ADAMS Accession No. ML080350347), Cooper Nuclear Station on June 30, 2008 (ADAMS Accession No. ML081540280) and for Davis Besse Nuclear Power Station, Unit 1 on June 30, 2008 (ADAMS Accession No. ML081410652).

## 2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called rated thermal power (RTP). Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K 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 emergency core cooling system (ECCS) analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the analyses. In practice, many of the design bases analyses assume a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A revision to 10 CFR Part 50, Appendix K effective on July 31, 2000, allows licensees to use a power level less than 1.02 times the RTP, but not less than the licensed power level, based on the use of state-of-the art feedwater flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. Because 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, the final rule by itself did not allow increases in licensed power levels. Because the licensed power level for a plant is a TS limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. North Anna Units 1 and 2 were initially licensed to operate at a maximum of 2775 MWt. In Amendments 84 and 71, dated August 25, 1986, the NRC approved North Anna Units 1 and 2 operation at the current maximum power level of 2893 MWt which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR Part 50, Appendix K. Currently, with the RTP of 2893 MWt, an analytical power level of 2951 MWt (102-percent of 2893 MWt) is used in the safety analysis. With a requested revised RTP of 2940 MWt and a revised uncertainty, the analytical power level is unchanged at 2951 MWt.

The desired MUR PU will be accomplished by increasing the electrical demand on the turbine-generator. As a result of this demand increase, steam flow will increase and the resultant steam pressure will decrease. The reactor coolant system (RCS) nominal cold leg temperature will decrease slightly while the hot leg temperature will increase slightly in response to the increased steam flow demand. As a result, the RCS average temperature will remain constant.

The NRC staff finds that the LEFM-assisted core thermal power measurement uncertainty is limited to 0.36 percent of actual reactor thermal power and, therefore, can support the proposed 1.6 percent PU. This results in the proposed increase of 1.6 percent in the North Anna Units 1 and 2 license power level using current NRC-approved methodologies.

## 3.0 TECHNICAL EVALUATION

### 3.1 Human Factors

#### 3.1.1 Regulatory Evaluation

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

#### 3.1.2 Technical Evaluation

The NRC staff has developed a standard set of questions for human factors reviews (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4) of MURs. The following sections evaluate the licensee's response to these questions in the license amendment request (LAR).

##### 3.1.2.1 Operator Actions

The licensee stated in its submittal that the existing required operator actions are not affected by the MUR PU. The licensee also stated that there is no reduction in the time required for the necessary operator actions. The licensee also determined that there were no new manual actions or automation of existing operator actions required as a result of the MUR.

The NRC staff has reviewed the licensee's statements in their submittal relating to any impacts of the MUR PU to existing or new operator actions credited in the safety analyses. The NRC staff concludes that the proposed MUR PU will not adversely impact operator actions and their response times because there were no changes identified. The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.1 of Attachment 1 to RIS 2002-03.

##### 3.1.2.2 Emergency and Abnormal Operating Procedures

The licensee stated in its submittal that the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) were reviewed for potential changes due to the proposed MUR PU. The licensee concluded that no changes to operator actions are required due to the MUR PU. The licensee identified EOP setpoints that require revisions. These setpoints will be revised to reflect the new total core power. The licensee stated that the affected procedures will be revised using the data generated with decay heats at the MUR power level. The licensee states that operator training on the procedure changes will be provided as a part of the MUR implementation.

The NRC staff concludes that the proposed MUR PU does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon: (1) The licensee making revisions to the EOPs and AOPs that will reflect the new power level and revised setpoints, and (2) the minor changes being made to the EOPs and AOPs will be reflected in the operator training program prior to MUR implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.A, VII.3, and VII.4 of Attachment 1 to RIS 2002-03.

3.1.2.3 Changes to Control Room Controls, Displays, and Alarms

In its submittal, the licensee described changes to control room controls, displays including the Safety Parameter Display System (SPDS), and alarms related to the proposed MUR PU. Notable proposed changes to controls, displays, and alarms include:

1. Instruments associated with turbine first stage pressure will require scaling changes
2. Instrument loops are affected by the PU (indicator replacement, calibration span, and/or scaling)
3. Plant computer points will be added and/or changed for the revised calorimetric algorithm and the feedwater ultrasonic flow meter
4. The new Ultrasonic Flow Meter (UFM) electronic cabinet, located in the Cable Spreading Room, is used to display and control aspects of feedwater flow data. The display provides system status or monitored process parameters. The display is typically used for maintenance purposes and not for control of plant operation
5. The system alerts operations personnel of UFM trouble through main control room overhead annunciator "Feedwater Ultrasonic Flow Meter Trouble." The main control room overhead annunciator "Feedwater Ultrasonic Flow Meter Failure" alerts the operators when the system loses a plane of operation, suffers a loss of AC power or other total failure. Any UFM condition that increases feedwater flow uncertainty is considered a "Feedwater Ultrasonic Flow Meter Failure" alarm condition.

All changes to the control room, including modifications involving the SPDS, will be reflected in the operator training program prior to MUR implementation.

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

3.1.2.4 Control Room Plant Simulator and Operator Training

The licensee stated that potential simulator changes will be identified as part of the plant modification process. The licensee also stated that these modifications will be evaluated, implemented and tested per the approved plant procedures. The licensee stated that simulator fidelity will be revalidated using approved plant procedures. The licensee stated that these modifications will be completed in time to support operator training prior to the MUR PU implementation.

The licensee stated in it's submittal that operator training will be developed and the operations staff will be trained on the plant modifications, Technical Specifications, technical requirements manual, and procedural changes prior to MUR PU implantation. The licensee stated that these changes, along with the plant simulator modifications, will be made prior to MUR implementation.

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

### 3.1.3 Conclusion

The NRC staff has completed its human factors review of the licensee's proposed changes and concludes that the licensee has adequately considered the impact of the proposed MUR PU on operator actions, EOPs and AOPs, control room components, the plant simulator and operator training programs.

## 3.2 Dose Consequences Analysis

### 3.2.1 Regulatory Evaluation

Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," recommends that to improve efficiency of the staff's review, licensees requesting an MUR uprate should identify existing design basis accident (DBA) analyses of record which bound plant operation at the proposed uprate power level. For any existing DBA analyses of record that do not bound the proposed uprate power level, the licensee should provide a detailed discussion of the reanalysis.

The NRC staff evaluated the impact of the proposed changes on analyzed DBA radiological consequences. Previously, in Amendments 240 and 221 dated June 15, 2005 (ADAMS Accession No. ML051590510), the licensee was granted implementation of a full-scope alternative source term (AST) in accordance with Title 10 of the *Code of Federal Regulations,* Part 50, Section 50.67 (10 CFR 50.67), and following the guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Therefore, this evaluation has been conducted to verify that the results of the licensee's DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67. Except where the licensee proposed a suitable alternative, the staff utilized the regulatory guidance provided in applicable sections of RG 1.183 and NUREG-0800, Standard Review Plan (SRP), Chapter 11, "Gaseous Waste Management System," SRP Chapter 15, "Transient and Accident Analyses," and SRP Chapter 6.4, "Control Room Habitability," in performing this review.

### 3.2.2 Technical Evaluation

The NRC staff reviewed the information related to the radiological consequences of DBA analyses provided by the licensee in Attachment 5 to the March 26, 2009, submittal. The findings of this evaluation are based solely on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee. The NRC staff reviewed the impact of the proposed 1.6 percent MUR PU on DBA radiological consequence analyses, as documented in

Chapter 15 of the North Anna Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). The specific DBA analyses that were reviewed were as follows:

- Loss-of-Coolant Accident (LOCA) Dose Evaluation
- Locked Reactor Coolant Pump (RCP) Dose Evaluation
- Fuel-Handling Accident (FHA) Dose Evaluation
- Main Steam Line Break (MSLB) Dose Evaluation
- Steam Generator Tube Rupture (SGTR) Dose Evaluation
- Volume Control Tank Rupture Dose Evaluation
- Waste Gas Decay Tank Rupture Dose Evaluation

The current DBA LOCA and the locked RCP dose analysis of record are based on a core inventory that assumes a power level of 2958 MWt, or 102.2-percent of the currently licensed thermal power of 2893 MWt. The steam release model for the locked RCP dose analysis assumes a core power level of 2951 MWt, or 102 percent of the currently licensed thermal power of 2893 MWt. Therefore, the current LOCA and locked RCP dose analyses remain bounding at the proposed MUR uprated power level of 2940 MWt with at least a 0.37-percent margin which bounds the 0.36-percent uncertainty associated with the LEFM CheckPlus™ system.

The current DBA FHA dose analysis of record for North Anna Units 1 and 2 is based on a core inventory that assumes a power level of 2958 MWt, or 102.2 percent of the currently licensed thermal power of 2893 MWt. The licensee made no changes to the assumed number of failed fuel rods or assumed radial peaking factor in support of the MUR PU. In addition, the licensee will maintain the continued applicability of the gap fractions for non-LOCA events as described in RG 1.183, Table 3, footnote 11. Therefore, the current FHA analyses remain bounding at the proposed power level of 2940 MWt with a margin, which bounds the 0.36-percent uncertainty associated with the LEFM CheckPlus™ system.

The current DBA, MSLB, and SGTR dose analyses of record for North Anna Units 1 and 2 are based on coolant activity levels set at the technical specification limits and as such the radiological source term for these analyses are independent of the RTP. The steam releases modeled for the MSLB and SGTR dose analyses assume a core power level of 2951 MWt, or 102-percent of the currently licensed thermal power of 2893 MWt. The licensee made no changes to any of the factors that affect the MSLB or SGTR dose consequence analyses. Therefore, the current MSLB and SGTR dose consequence analyses remain bounding at the proposed MUR uprated power level of 2940 MWt with margins that bound the 0.36-percent uncertainty associated with the LEFM CheckPlus™ system.

In a letter dated July 8, 2009, the licensee provided additional information describing the independence of RTP on the Waste Gas Decay Tank Rupture Dose Evaluation. The licensee has in place a program to control the quantity of radioactivity in each gas storage tank such that, in the event of an uncontrolled release of the tanks' contents, the whole-body exposure to any individual in an unrestricted area would not exceed 0.5 rem. Section 5.5.11 of the TS, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," states that, "This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." Section 5.5.11, Item b states that "A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of $\leq$0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents...."

The licensee's Explosive Gas and Storage Tank Radioactivity Monitoring Program meets the dose guidelines described in SRP Chapter 11. Therefore the staff concludes that the proposed MUR will have no impact on the Waste Gas Decay Tank Rupture Dose Evaluation since the amount of radioactivity in the tank is controlled to meet the 0.5 rem dose guideline independent of the RTP.

The current DBA, Volume Control Tank Rupture dose analysis of record for North Anna Units I and 2 is based on the guidance in Branch Technical Position (BTP) ETSB 11-5. The analysis assumes a source term corresponding to 1% failed fuel fission product inventory in the RCS. Based on the conservative gap activity used in the analysis, the MUR power uprate impact on the VCT rupture radiological consequences is bounded. Therefore, the current VCT dose consequence analyses remain bounding at the proposed MUR uprated power level of 2940 MWt with margins that bound the 0.36-percent uncertainty associated with the LEFM CheckPlusTM system.

Using the licensing basis documentation, as contained in the current UFSAR and TS, in addition to information in the March 26, 2009, LAR submittal letter, the staff verified that the existing UFSAR Chapter 15 radiological analyses source term and release assumptions bound the conditions for the proposed 1.6-percent PU to 2940 MWt, considering the higher accuracy of the proposed feedwater flow measurement instrumentation.

### 3.2.3 Conclusion

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to reassess the radiological consequences of the postulated DBA with the proposed uprate power level. The staff finds that the licensee will continue to meet the applicable dose acceptance criteria following implementation of the proposed 1.6-percent MUR PU. The staff further finds reasonable assurance that North Anna Units 1 and 2, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of the design-basis accidents.

## 3.3 Fire Protection

### 3.3.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe-shutdown of the plant are protected from the effects of a fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shut down the plant; (2) GDC 3 of Appendix A to 10 CFR Part 50, insofar as it requires that [a] SSCs important to safety be designed and located to minimize the probability and effect of fires, [b] noncombustible and heat resistant materials be used, and [c] fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

3.3.2 Technical Evaluation

The NRC staff has reviewed Attachment 5, Section II.2.36, "Safe Shutdown Fire Analysis (Appendix R Report)-UFSAR 9.5.1," and Section VII.6.A, "Fire Protection Program," of the LAR. The staff also reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection," (i.e., approved fire protection program). The review covered the impact of the proposed MUR PU 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 PU on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

By supplemental letter dated July 8, 2009, the licensee provided requested additional information. In its response, the licensee stated that the Appendix R fire safe-shutdown analyses were reviewed for the MUR PU and concludes: (1) existing operator actions are not affected; (2) no reduction in operator action time was identified; (3) no new operator actions were identified; and (4) no existing manual actions were automated. The temperature and pressure parameters outside of containment will not be impacted by implementation of the MUR PU. Therefore, Appendix R manual actions outside containment are not affected by additional heat in the plant environment from the increased power. Further, the licensee stated that there are no manual actions required inside containment to maintain safe-shutdown. There is only one manual action listed as a contingency (Fire Contingency Action Procedure 1/2-SCA-5) in the event that less than one residual heat removal (RHR) pump is available -- repair of RHR cable. This manual action would be required for cold shutdown approximately 55 hours after the fire initiating event. The extinguishment and hot shutdown time is approximately one hour. Therefore, given this very long available time margin (at least 50 hours), this action inside containment is not affected by additional heat in the plant environment from the increased power. The licensee concluded that any effects from additional heat in the plant environment from the increased power will not interfere with existing operator manual actions being performed at their designated time and place.

For the MUR PU condition, the licensee indicated that the proposed MUR PU does not impact the previous operator manual action times in the fire safe-shutdown analysis. The licensee indicated that there is only one manual action and it is required for cold shutdown repairs following a fire. This cold shutdown manual action time is 55 hours after the fire event. The licensee indicated that fire extinguishment activities and hot shutdown time is one hour. With consideration for increased heat in plant environment Appendix R manual actions can be still be accomplished within the time requirements.

In a letter dated June 17, 2009, the staff requested the licensee to verify whether the North Anna Units 1 and 2 credit aspects of their fire protection systems for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If the North Anna Units 1 and 2 credit their fire protection systems for other than fire protection activities, the MUR PU LAR should identify the specific situations and discuss to what extent, if any, the MUR PU affects these non-fire protection aspects of the plant fire protection system. If the North Anna Units 1 and 2 do not take such credit, the staff requested that the licensee verify this as well.

By letter dated July 8, 2009, the licensee provided additional information outlined in this paragraph in response to the staff's RAI. In its response, the licensee stated that the North Anna Units 1 and 2 fire protection systems are not credited or required to mitigate the consequences of Design-Basis Accidents. However, as noted in UFSAR Chapter 9.5, in addition to its primary function, which is to permit safe-shutdown of the plant in the event of a fire, the fire protection system also provides alternate sources of make-up water for the spent fuel pool and for the Unit 1 and 2 auxiliary feedwater systems. The licensee stated that in accordance with BTP APCSB 9.5-1, Appendix A, Paragraph A.4, postulated fires need not be considered concurrently with other plant accidents.

The licensee concluded that these secondary functions of the fire protection system do not prohibit the system from performing its primary functions. The fire protection system's capacity remains adequate to provide secondary functions (i.e., backup water to auxiliary feedwater pumps or make-up water to the spent fuel pool) at the uprated power. In addition, the fire protection system is used in B.5.b mitigating strategies as a method to supply make-up water to various systems. Furthermore, the licensee concluded that the MUR has no impact on the ability of the fire protection system to meet the make-up requirements of the mitigating strategies.

The licensee indicated that the fire protection system is credited as an alternate source for make-up water to the spent fuel pool and auxiliary feedwater systems. For this make-up activity, because postulated fires are not considered concurrent with other plant accidents thereby not requiring concurrent primary and secondary demands on the fire protection system, and because the secondary functions of the fire protection system are not affected by the MUR PU, the staff concluded that the MUR PU has no effect on the "non-fire protection" aspects of the plant fire protection system. The licensee also stated that the fire protection system is used in B.5.b mitigating strategies to supply make-up water to various systems and concluded that the MUR PU has no impact on make-up requirements. Based on the licensee's response, the staff concluded that the MUR PU has no impact on the ability of the fire protection system to meet the make-up requirements of the above mitigating strategies.

### 3.3.3 Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the 1.6-percent increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The staff finds this aspect of the capability of the associated SSCs to perform their design-basis functions at an increased core power level of 2940 MWt acceptable.

## 3.4 Chemical Engineering

### 3.4.1 Flow-Accelerated Corrosion (FAC)

#### 3.4.1.1 Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, flexibility to control these parameters to minimize FAC is limited. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed MUR PU on FAC and the adequacy of the licensee's FAC program. The intent of the FAC program is to predict the rate of loss so that repair or replacement of damaged components can be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989, NRC Generic Letter 89-08 (GL 89-08), "Erosion/Corrosion - Induced Pipe Wall Thinning," May 1989, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," April 1999. It consists of predicting loss of material using the CHECWORKS steam or feedwater application (SFA) FAC monitoring computer code, visual inspection, and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

#### 3.4.1.2 Technical Evaluation

The licensee stated that the FAC program implemented at North Anna Units 1 and 2 provides a standardized method of identifying, inspecting, and tracking components susceptible to FAC wear in both single-and two-phase flow conditions. The licensee utilizes the CHECWORKS SFA FAC monitoring computer code to predict and track FAC susceptible components.

The CHECWORKS SFA computer code is used to create unit-specific databases; whereby, analysis and data interpretation is performed. The licensee stated that the analytical models developed from CHECWORKS SFA is used to determine wear-rate analyses that rank components in order of predicted wear and time to reach minimum code wall thickness. In support of the MUR PU, the licensee stated that the CHECKWORKS SFA FAC monitoring computer code was updated to incorporate the changes associated with the power uprate. The associated changes include: thermal power, pressure, temperature and flowrate.

The licensee stated that an evaluation was performed to determine the impact the increase in wear rates, due to the MUR PU, had on remaining service life of systems affected by FAC. The licensee indicated that upon PU implementation, the CHECKWORKS SFA databases for North Anna Units 1 and 2 will be updated and validated. The licensee stated that the key design parameters for the systems monitored by FAC are reactor coolant system (RCS) temperature secondary-side steam temperature and pressure, and feedwater temperature. Further, the licensee stated that upon power uprate implementation, the wear rate analysis model will be analyzed using the updated information obtained from inspections and evaluations of wear rates from plant systems susceptible to FAC. In addition, it was indicated that the components with a low or negative time for remaining service life will be evaluated for re-inspection.

The licensee identified piping components in the five systems expected to experience the greatest increase in FAC wear as a result of the MUR PU. The piping components that are expected to experience the greatest increase in FAC wear in Unit 1 are part of the steam drain system; in Unit 2, they are part of the extraction steam system. The increase in predicted FAC wear rate associated with the piping components mentioned above is manageable under the FAC program and is thus acceptable. The licensee performed wear rate analyses and have concluded that increased wear rates on remaining service life is not significant. The staff reviewed the wear rate analyses results and concurs with the licensee's conclusion. It was further stated that the remaining service life for the FAC susceptible service lines will continue to be monitored and will be documented at the end of each refueling outage.

The staff has reviewed the licensee's evaluation and confirms that the applicable regulatory guidance was followed. The staff has also verified the calculations in the application. The licensee has demonstrated that the FAC program is adequate for managing the potential effects on the piping components susceptible to FAC. The staff's acceptance of the licensee's use of CHECKWORKS can be found in NUREG-1766, "Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2," December 2002. The staff concurs that the FAC program is adequate in predicting the rate of material loss.

3.4.1.3 Conclusion

On the basis of its review, the NRC staff concludes that the FAC program is acceptable for MUR PU operating conditions because the effect on FAC rates adequately addressed the impact of changes in the plant operating conditions on the FAC analysis. The licensee has demonstrated that the updated analyses will predict the loss of material by FAC, and allow for timely repair or replacement of degraded components following implementation of the proposed MUR PU. Therefore, the staff finds the proposed LAR acceptable with respect to the FAC program.

3.4.2. Coatings

3.4.2.1 Regulatory Evaluation

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and radionuclide contamination and provide wear protection during plant operation and maintenance activities. The coatings are subject to 10 CFR Part 50, Appendix B. The NRC staff review focused on whether the pressure and temperature conditions under the proposed power uprate continue to be bounded by the conditions to which the coatings were qualified.

3.4.2.2 Technical Evaluation

The licensee stated that the coating systems used inside containment are used to protect equipment and structures from corrosion and radionuclide contamination. It was further stated that the coatings also provide wear protection during plant operation and maintenance activities. The containment service level 1 coatings were originally qualified to withstand a loss-of-coolant accident (LOCA) environment and meet American National Standards Institute (ANSI) Standards

N5.12, “Protective Coatings (Paints) for the Nuclear Industry”, N101.2, “Protective Coatings for Light Water Nuclear Reactor Containment Facilities,” and N101.4, “Quality Assurance for Protective Coatings Applied to Nuclear Facilities.” The license stated that there were no changes to the containment analyses that would require a change to the containment design pressure and temperature. Therefore, since the containment design pressure and temperature limits are not changing, the licensee concluded that the service level 1 containment coatings remain qualified for operation under PU conditions. The staff concurs that the coatings will not be adversely impacted by the MUR PU and that temperature and pressure limits under PU conditions continue to be bounded by the conditions to which the coatings were qualified.

3.4.2.3 Conclusion

On the basis of its review, the NRC staff concludes that the coatings will not be adversely impacted by the MUR PU. The temperature and pressure limits under PU conditions continue to be bounded by the conditions to which the coatings were qualified. Therefore, the staff finds the proposed MUR PU acceptable with respect to coatings.

3.4.3 Steam Generator Program

3.4.3.1 Regulatory Evaluation

Steam generator (SG) tubes constitute a significant part of the reactor coolant pressure boundary (RCPB). SG tube inservice inspection provides a means for assessing the structural and leak-tight integrity of SG tubes through periodic inspection and testing of critical areas and features of the tubes. The staff reviewed the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed power uprate on the design and operation of SGs. Specifically, the staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis; in other words, the technical specification (TS) plugging limits.

3.4.3.2 Technical Evaluation

North Anna Units 1 and 2 each has three Westinghouse Model 54F replacement SGs, which were replaced in 1993 and 1995, respectively. Each SG contains 3592 thermally treated Alloy 690 tubes. The tubes have an outside diameter 0.875 inches, a wall thickness of 0.050 inches, and are supported by seven stainless steel tube support plates and a baffle plate. Most of the tube support plate holes are quatrefoil-shaped. The U-bend region of the tubes in rows one through eight was stress relieved after bending.

Potential degradation mechanisms for North Anna Units 1 and 2 SG tubes include: (1) tube support plate wear, (2) anti-vibration bar (AVB) wear, (3) loose parts wear, (4) outside diameter stress corrosion cracking (ODSCC), (5) primary water stress corrosion cracking (PWSCC), and (6) pitting. Although, the licensee indicated that AVB wear, loose parts wear, PWSCC, and pitting have not been identified in the North Anna Units 1 and 2 current SGs, the degradation mechanisms are included in the inspection planning. Corrosion mechanisms such as ODSCC, PWSCC, and pitting are influenced by an increase in operating temperature and affected by chemistry changes. Mechanical processes such as AVB wear and loose parts wear are more dependent on tube bundle flow rates.

The licensee reported that the North Anna Units 1 and 2 Model 54F SGs have had no incidence of PWSCC because thermally treated Alloy 690 tubes have high resistance to this degradation mechanism. In addition, the licensee stated that the increased RCS temperature will not significantly change the mechanical wear processes and PWSCC. It was further stated that the chemistry effects on tube corrosion after PU are insignificant.

The licensee performed a SG tube wear evaluation based on the current design-basis analysis and consideration of SG secondary side thermal-hydraulic changes resulting from the MUR PU. SG tube wear due to fluid-elastic effects in the U-bend region and turbulence-induced displacement effects in the straight leg tube region were considered. The calculated tube stability ratio and the baseline vibration amplitude were found to be below the licensee's acceptance criteria, including the tube-to-tube gap width. The licensee determined that the turbulence effects are acceptable. The staff concurs with the licensee's evaluation.

Based on the SG tube wear evaluation, the licensee stated that the maximum SG baseline tube wear is small, 0.008 inches over the 60 years of plant operation. The licensee determined that the revised wear is projected to increase from the pre-uprate value of 0.008 inches to approximately 0.012 inches post uprate. The licensee determined that this amount of tube wear will not affect tube integrity, and, is therefore, acceptable. It was further indicated that the tube repair limit is unaffected by the MUR PU and remains valid at uprate conditions. The staff concurs that this will not be adversely impacted by the MUR PU.

The staff has reviewed the licensee's evaluation and calculation results and has confirmed that the applicable regulatory guidance was followed. The staff concurs that the proposed PU will introduce only insignificant changes as it relates to tube wear due to potential tube degradation mechanisms, which will not affect satisfactory performance in maintaining SG tube integrity.

#### 3.4.3.3 Conclusion

On the basis of its review, the NRC staff concludes that the licensee has adequately assessed the continued acceptability of the plant's TSs. Specifically, the licensee has an ongoing inspection program, including periodic degradation assessments, that will identify any increased degradation and any need for additional or enhanced inspections prior to tube integrity being challenged. Therefore, the staff finds the proposed MUR PU acceptable with respect to the SG tube inservice inspection.

### 3.4.4 Chemical and Volume Control System (CVCS)

#### 3.4.4.1 Regulatory Evaluation

The chemical and volume control system (CVCS) provides a means for: (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary-water chemistry and reducing coolant radioactivity level, and (5) supplying recycled coolant for demineralized water make-up for normal operation and high-pressure injection flow to the emergency core cooling system (ECCS) in the event of postulated accidents. The staff has reviewed the safety-related functional performance characteristics of CVCS components. The NRC's acceptance criteria are based on (1) GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," 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; and (2) GDC-29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in Standard Review Plan (SRP) Section 9.3.4, "Chemical and Volume Control System (PWR)."

3.4.4.2 Technical Evaluation

The licensee stated that the major plant parameters for the nuclear steam supply system (NSSS) design transients are RCS hot- and cold-leg temperatures, secondary-side steam temperature and pressure, and the secondary-side feedwater temperature. It was further stated that the existing design transients for the parameters, except for feedwater temperature, bound plant operation at the uprated power conditions. The uprated full power feedwater temperature has been analyzed to increase 9 degrees Fahrenheit (°F) from its current design value of 440°F. The licensee revised the design transients with the uprated feedwater temperature. In addition, the licensee revised previous design analyses by replacing the generic transients with more system-specific analyses to better reflect uprate plant conditions. The licensee reviewed analyses results and determined that the fatigue analysis for the various NSSS components remained satisfactory. The primary-to-secondary differential pressure limit was not exceeded for any normal or upset design transients. Additionally, the licensee indicated that no new design transients are created as a result of the MUR PU. The staff concurs this will not be adversely impacted by the MUR PU.

In a response letter dated July 8, 2009, the licensee provided additional information regarding nitrogen-16 activity and its relationship with letdown line decay time requirements. The letdown line decay requirement does not depend on reactor power, but rather, depends on letdown flowrate and total volume of piping through which flow passes between the RCS loop and the containment penetration. The licensee stated that the MUR power uprate does not affect letdown flowrate or letdown piping volume; and therefore, the design requirements remain satisfied for operation at the proposed MUR power uprate conditions. The staff found the response to the RAI acceptable.

The staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The licensee has demonstrated that the CVCS will continue to maintain RCS inventory and water chemistry. The staff concurs that the CVCS will continue to meet system design requirements and that no knew design transients will be created at MUR PU conditions.

3.4.4.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the CVCS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and its effects on the CVCS. The staff further concludes that the licensee has demonstrated that the CVCS will continue to be acceptable and will continue to meet the requirements of GDC 14 and GDC 29 following implementation of the proposed MUR PU.

## 3.4.5 Steam Generator Blowdown System (SGBS)

### 3.4.5.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The 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 design flows for all modes of operation. The NRC staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criteria for the SGBS are based on GDC 14 which requires the RCPB to be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

### 3.4.5.2 Technical Evaluation

The licensee stated that the SGBS flow rates during plant operation are based on chemistry control and tubesheet sweep necessary to control solids buildup. The licensee stated that the SGBS flowrate will continue to be operated with no change under MUR PU conditions. It was further stated that SGBS operating temperatures and pressures will decrease, but remain bounded by the existing design parameters under uprate conditions. The licensee has determined that the SGBS will continue to meet system design requirements at the proposed PU conditions. The licensee stated that the uprate will not significantly increase the potential for FAC on the blowdown system piping and components. The SGBS system will continue to be monitored by an acceptable FAC monitoring program. The staff finds this acceptable.

The staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The licensee has demonstrated that the SGBS is adequate for maintaining secondary-side water chemistry within industry guidelines for maintenance of controlled corrosion rates in secondary system components. The staff concurs that the SGBS will continue to meet system design requirements at MUR PU conditions.

### 3.4.5.3 Conclusion

The staff has reviewed the licensee's evaluation of the effects of the proposed MUR PU on the SGBS and concludes that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The staff further concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of GDC 14 following implementation of the proposed MUR PU.

## 3.4.6 Overall Chemical Engineering Conclusion

In the areas of SGs and chemical engineering, the NRC staff concludes that the licensee has adequately addressed (1) the changes in the plant operating conditions for the FAC program, (2) the effects on protective coatings, (3) the changes in the SG operating parameters, the effects on the SGs and the determination that the SG tube integrity will continue to be maintained, (4) the changes of the reactor coolant and their effect on the CVCS, and (5) the changes in the system flow and impurity levels, and their effects on the SGBS.

## 3.5 Mechanical and Civil Engineering

### 3.5.1 Regulatory Evaluation

The NRC staff's review in the areas of mechanical and civil engineering covers the structural and pressure boundary integrity of NSSS and balance-of-plant (BOP) systems and components. This review focuses on the impact of the proposed MUR PU on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor pressure vessel (RPV) and its supports; (4) the pressure retaining portions of the control rod drive mechanisms (CRDMs); (5) the replacement steam generators (RSGs) and their supports; (6) the pressure retaining portions of the reactor coolant pumps (RCPs) and their supports; (7) the pressurizer and its supports, and (8) the reactor vessel internals (RVIs). Technical areas covered by this review include stresses, cumulative usage factors (CUFs), flow-induced vibration (FIV), high-energy line break (HELB) locations, and jet impingement and thrust forces.

The primary guidance used by the licensee and other licensees for LARs involving MUR PUs is found in Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," which provides licensees with a guideline for organizing LAR submittals for MUR PUs. 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 aforementioned components and technical areas.

The above affected piping systems, components and their supports, including core support structures, are designed in accordance with the rules of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section III, the United States of America Standards (USAS)/American National Standards Institute (ANSI) B31.1 Code for Power Piping, and the USAS/ANSI B31.7 Code for Nuclear Power Piping. The NRC staff's evaluation considered General Design Criteria (GDC) 1, 2, 4, 14, and 15 which are located in 10 CFR Part 50, Appendix A. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR Part 50, Section 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.

3.5.2 Technical Evaluation

The review focused on the effects of the PU on the structural and pressure boundary integrity of piping systems and components, their supports, the reactor vessel and internal components, the CRDMs, and the BOP and NSSS interface piping systems. The PU will be achieved by an increase in demand to the turbine generator. In turn, an increase in steam flow will occur due to the increased demand on the secondary side of the plant. In addition, there will be an increased temperature difference across the core with the reactor coolant system (RCS) pressure remaining the same. Table 4.0-2 of Attachment 1 of the LAR shows the pertinent temperatures, pressures, and flow rates for the current and proposed uprate conditions. For the purposes of the evaluation in support of the proposed power uprate, the licensee utilized the most bounding case (Case 4) found in Table 4.0-2. At full power, the hot-leg temperature increases from 621.2°F to 621.9°F, while the cold-leg temperature decreases from 552.3°F to 551.7°F. The RSG pressures increase slightly from 850.0 psia to 851.0 psia and the steam flow increases from 12.78 million pounds per hour (Mlbm/hr) to 13.19 Mlbm/hr. The design feedwater (FW) temperature increases from 440.0°F to 449.0°F. The design parameters for the primary system at North Anna Units 1 and 2 are found in Chapter 5 of the North Anna Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). The RCS components are designed to 650°F (except the pressurizer and pressurizer surge line, which are designed to 680°F) and 2,485 psig. Chapter 10 of the North Anna Units 1 and 2 UFSAR provides the design-basis information for the secondary side systems including the main steam (MS) system and the FW and Condensate system.

The proposed uprate does not change heatup or cooldown rates or the number of cycles assumed in the design analyses. In addition, the limiting analyses for design transients are still bounding.

3.5.2.1 Reactor Pressure Vessel and Internals

The code of record for the RPV, nozzles, and supports is the ASME Code, Section III, 1968 Edition through the winter 1968 Addenda. The code of record for the replacement reactor vessel closure heads (RVCHs) at North Anna Units 1 and 2 is the ASME Code, Section III, 1995 Edition with 1996 Addenda. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analyses of record. The licensee confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. It is noted in Section IV.1.A.i in Attachment 5 of LAR that the maximum vessel outlet temperature increase of 0.7°F slightly increases the severity of the plant loading and unloading transients at the main closure flange assemblies, CRDM housings, and the vessel outlet nozzles. However, these also remain bounded by the current design basis analyses for these components. Also, no additional transients have been proposed as a result of the MUR uprate at North Anna Units 1 and 2.

With regards to the reactor vessel internals (RVIs), the licensee stated that mechanical and structural evaluations were performed on these components to determine any effects on the RVIs due to the uprated conditions. The PU will not affect the design basis for the seismic and LOCA loads for North Anna Units 1 and 2 negating the need to re-assess the structural integrity of the RVIs with regards to the LOCA-induced hydraulic and dynamic loads and seismic loads. Additionally, the licensee reviewed the possible effects of flow induced vibration (FIV) on the reactor vessel internals to ensure that fatigue endurance limits for the components under review would not be approached due to the uprated conditions. It was confirmed by the licensee that the portions of the internals most vulnerable to FIV, the core barrel assembly and upper internals, contained adequate structural margin such that FIV would have a negligible impact on the structural integrity of these components. Structural evaluations performed by the licensee demonstrated that all but two components making up the reactor internals would be bounded by current design-basis analyses at the uprated conditions. For these components, the lower and upper core plates, the licensee performed additional confirmatory structural analyses to demonstrate that they would maintain their structural integrity at the uprated conditions. For these components, the lower and upper core plates, the licensee performed additional confirmatory structural analyses to demonstrate that they would maintain their structural integrity at the uprated conditions. Given that the lower core plate is affected by the higher heat generation rates produced by the power increase, the licensee accounted for this increase in comparing the maximum primary plus secondary stress intensity and the cumulative usage factor (CUF) of this component at the uprated conditions to its respective code allowable value (see Section 4.2.2 of NAPS UFSAR for reactor internals design code requirements), and demonstrated that the lower core plate is structurally adequate for operation at the uprated condition. Similarly, for the upper core plate, the licensee accounted for the increased average temperature difference experienced by the upper core plate and also compared the maximum primary plus secondary stress intensity and CUF at the uprated conditions of the upper core plate to its respective code allowable value (also found in Section 4.2.2 of NAPS UFSAR), and demonstrated that the upper core plate is structurally adequate for operation at the uprated condition.

As indicated, the existing loads, stresses, and fatigue CUF values for the RPV and internals remain valid for the proposed power uprate at North Anna Units 1 and 2. The NRC staff concurs with the licensee's assessment that the RPV and internals are acceptable for operation at the uprated power level given that the current design-basis analyses remain valid for these components at the uprated conditions.

3.5.2.2 Control Rod Drive Mechanisms

The code of record for the pressure retaining components of the CRDMs is the ASME Code, Section III, 1968 Edition through the Winter 1969 Addenda. Based on the location of the CRDMs, these components are affected primarily by the RCS pressure and the RCS hot leg temperature of which only the hot leg temperature changes as a result of the power uprate (0.7°F increase to 621.9°F). Based on the fact that a maximum hot leg temperature of 622°F was used in the analysis of record for the CRDMs, the licensee confirmed in its submittal that the CRDM design analyses continue to bound the conditions at the proposed uprated power level and all critical margins on these components will be maintained. In addition, the operating hot leg transients continue to bound the uprate conditions and no additional transients have been proposed. The NRC staff concurs with the licensee's assessment that the CRDMs are acceptable for operation at the uprated power level due to the bounding nature of the current CRDM design analyses.

3.5.2.3 Reactor Coolant Piping and Components

The RCS piping was designed to ANSI/USAS B31.7 Code for Nuclear Power Piping, 1969 Edition with 1970 and 1971 Addenda. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the reactor coolant piping and supports. It was stated that there is no change in RCS design or operating pressure, and the effects of the variance in operating temperature for the hot leg (increases at uprated conditions) and cold legs (decreases at uprated conditions) are within design limits. The MUR PU conditions are bounded by the design conditions and the RCS piping remains within the allowable stress limits provided by the applicable codes of record identified above. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCS piping and supports remain valid for the proposed PU.

The RSGs were installed in 1993 and 1995 at Units 1 and 2, respectively. The new steam generators replaced the original tube bundles, lower shells, and primary channel head regions while the original upper shells (steam drums) remained. Based on the combination of the original and new portions of the RSGs, two editions of Section III of the ASME Code were used in the qualification of these components. The upper portions of the current RSGs (the steam drum vessels) which remain from the previous SGs were designed to the 1968 Edition with Winter 1968 Addenda, while the lower assembly portions of the SGs which were replaced at North Anna Units 1 and 2 were designed to the 1986 Edition of Section III of the ASME Code. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the various components which make up the RSGs. There is a no change in RCS mass flow rate and the RCS temperatures and pressures used in the design continue to bound the uprate conditions. With regards to the structural evaluation of the primary and secondary side components of the RSGs, the licensee focused primarily on the structural impact of the primary-to-secondary differential pressure which changes as a result of the uprated conditions. Scale factors reflecting the uprated conditions were utilized by the licensee in determining what effect the uprate would have on the component stresses within the RSGs. These stresses were compared to the ASME Code allowable values for the RSG components to determine the structural adequacy of these components at the uprated conditions. Most of the components of the RSGs were found to meet the applicable ASME Code allowable stress requirements. However, it was found by the licensee that the secondary manway bolts of the RSGs at North Anna Units 1 and 2 did not meet the code allowable stress value of 3Sm. A simplified elastic-plastic analysis was performed, in accordance with ASME Code requirements, to determine what effect the uprated conditions would have on the structural integrity of these bolts. From this analysis, the licensee concluded that the manway bolts meet the ASME Code allowable fatigue limits for a service period of "approximately 45 years." Per the summary of this analysis in Section IV.1.B.ii of Attachment 5 of the LAR, the licensee indicated that two courses of action could be taken regarding the service life of these components: 1) these bolts would have to be replaced at North Anna Units 1 and 2 prior to reaching 45 years of service or 2) additional analyses would need to be performed to demonstrate the continued use of these bolts past 45 years based on actual operating conditions during this service period. This is also reflected in Attachment 6, where the licensee has formally committed (Commitment Number 7) to take one of the two aforementioned courses of action for the RSG manway bolts. The existing loads, stresses, and fatigue CUF values for the components of the RSGs remain valid for the proposed MUR uprate, dependent on the aforementioned commitment.

The pressure retaining parts of the RCPs were designed in accordance with the ASME Code, Section III, 1968 Edition through the Winter 1970 Addenda. The licensee reviewed the revised design conditions to determine the impact on the existing design-basis analyses for the RCPs. Due to the arrangement of the North Anna Units 1 and 2 RCS, the RCP loading conditions and thermal transients are governed by the reactor vessel inlet temperature, i.e., the cold leg temperature. As previously mentioned, the change in the cold leg temperature corresponding to the uprated conditions is within current design limits at North Anna Units 1 and 2 since a vessel inlet temperature of 552.3°F was utilized in the analyses supporting the replacement of the steam generators. Given that higher temperatures are much more restrictive with regards to the stress requirements for the pressure retaining portions of the RCPs, it was concluded that the existing design-basis analyses and the existing loads, stresses, and fatigue CUF values for the RCPs remain valid and bounding for the MUR PU. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed.

The code of record for the pressurizer at North Anna Units 1 and 2, including the nozzles, is the ASME Code, Section III, 1968 Edition through the Winter 1968 Addenda. The current design code of record for the pressurizer surge line is provided as the 1986 Edition of Section III of the ASME Code (re-evaluated to address NRC Bulletin 88-11). With regards to the structural integrity of the pressurizer, the governing loading conditions occur when the RCS pressure is elevated and the hot and cold leg temperatures are low. The licensee reviewed the revised design conditions to determine the impact on the existing structural design-basis analyses for the pressurizer. Given that the RCS pressure remains unchanged at the uprated conditions and the hot and cold leg temperature variations at these conditions remain bounded, the licensee concluded that the current design-basis analyses remain bounding for the North Anna Units 1 and 2 pressurizers. An extensive discussion regarding the design transients utilized in the evaluation of the structural integrity of the pressurizers was provided in Section IV.1.A.viii of Attachment 5. The licensee confirmed that the existing stress and fatigue evaluations remain valid based on the fact that the NSSS design transients and insurge/outsurge transient effects are not affected by the PU.

Additionally, no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for pressurizer remain valid for the proposed PU.

The NRC staff concurs with the licensee's conclusion that the design of the reactor coolant piping and components, including the RSGs, RCPs, and pressurizer, and their supports, are adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop since the design-basis analyses of these components remain bounding for the proposed 1.6-percent PU condition. This conclusion is also dependent on the commitment outlined by the licensee regarding the RSG manway bolts.

3.5.2.4 High Energy Line Break Locations

The licensee stated that the current HELB analysis for North Anna Units 1 and 2 was reviewed in support of the proposed MUR PU. Section IV.1.B.vii of LAR states that the changes in the NSSS parameters (e.g. temperature, pressure, mass flow rates) were reviewed to determine whether the changes in these parameters would have a significant impact on the system pipe stresses. It was concluded that the pipe stresses were not significantly affected by the parameter changes resulting from the PU. In addition, no new piping was added, no postulated break locations were changed, no changes were made to the assumed blowdown from the current postulated break locations, and there are no new systems that qualify as HELB systems as a result of the uprate. Based on this information, the licensee concluded that the current North Anna Units 1 and 2 HELB analysis remains unaffected by the uprate. Additionally, the licensee indicated that the existing leak before break (LBB) analyses used to justify the elimination of large break pipe ruptures of the RCS loop piping at North Anna Units 1 and 2 remain valid for the uprated conditions. The NRC staff concurs with the licensee's conclusion regarding HELBs and the LBB analyses given that the current analyses remain bounding and there are no changes to the analyses required in support of the uprate.

3.5.2.5 BOP Piping Systems

The licensee evaluated the BOP piping systems by comparing the conditions for the proposed PU with the analyses of record conditions and the current operating conditions. In response to a staff RAI regarding these systems, the licensee provided a detailed listing of the BOP piping systems and NSSS interface systems which were evaluated in support of the MUR PU at North Anna Units 1 and 2. The BOP piping systems evaluated included the Main Steam and Steam Dump System, Moisture Separator & Reheater Drain System, Feedwater System, Condensate System, Extraction Steam System High Pressure Feedwater Heater Drains System, Low Pressure Feedwater Heater Drains System, and the Auxiliary Feedwater System, among others.

The licensee stated that change factors (i.e., ratios of pre-uprate conditions versus post-uprate conditions) were utilized to analyze the effect that the changes in pressure, temperature, and flow would have on the aforementioned piping systems after the proposed PU. If a change factor was greater than one (1) for any of the three parameters for a specific system, an evaluation was performed to determine whether the design-basis analyses for the system under review would remain bounding. It was stated that each of the BOP piping systems will remain bounded by their existing design-basis analyses following the evaluations performed utilizing the change factors. Therefore, it was concluded that the existing loads, stresses, and fatigue CUF values for the aforementioned piping systems remain valid for the proposed PU. Additionally, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. Based on the above review which confirms that these systems remain bounded by their existing design-basis analyses, the NRC staff concurs with the licensee's conclusion that the proposed 1.6-percent PU will not have adverse effects on BOP system piping.

3.5.2.6 Safety-Related Valves

The NRC staff reviewed the licensee's safety-related valves analysis. The evaluation showed that no changes in reactor coolant system design or operating pressure were made as part of the MUR PU, and the temperature changes due to MUR PU are bounded by those used in the existing analyses. The analyses also confirmed the installed capacities and lift setpoints of the main steam safety valves remain valid for the MUR PU conditions. Due to the insignificant changes in temperature and operating pressure, none of the safety-related valves required a change to their design or operation as a result of the MUR PU.

The licensee also stated that systems which have valves maintained within the air-operated valve (AOV) program, the GL 89-10 motor-operated valve (MOV) program, and the GL 95-07 pressure locking/thermal binding program were evaluated. The system evaluations for AOVs indicate that the MUR PU does not affect the maximum differential pressure, line pressures, flow rates, or fluid temperatures. The evaluations of GL 89-10 identified MOVs indicate that the maximum differential pressures, line pressures, and flow rates are not affected by the MUR PU. The evaluations of the valves in the GL 95-07 pressure locking/thermal binding program determined that the thrust required to open the applicable valves remains less than the motor actuator capabilities at MUR PU conditions. Based on the licensee's evaluations, the staff finds the performance of existing safety-related valves acceptable with respect to the MUR PU.

3.5.2.7 Safety-Related Pumps

The NRC staff reviewed the licensee's safety-related pumps analysis. The NRC's acceptance criteria for reviewing the safety-related pumps analysis are based on 10 CFR 50.55a. The licensee reviewed the impact of the proposed MUR PU conditions on the existing design-basis analyses for the safety-related pumps. The evaluation showed that there are no significant changes to the maximum operating conditions and no changes to the design-basis requirements that would affect pump performance or the acceptable test criteria for the pumps. Therefore, the staff finds the performance of existing safety-related pumps acceptable with respect to the MUR PU.

3.5.2.8 Inservice Testing (IST) program

In its submittal, the licensee described its review of the IST Program for safety-related pumps and valves at North Anna Units 1 and 2 for MUR operations. The Code of Record for North Anna Units 1 and 2 is the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 1995 Edition through the 1996 Addenda. The IST Program at North Anna Units 1 and 2 assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. The scope of the IST program must be periodically updated to meet applicable ASME OM Code requirements specified in 10 CFR 50.55a. However, in support of the MUR request, no effects are anticipated in the IST Program at North Anna Units 1 and 2.

3.5.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR PU on NSSS and BOP systems and components with regard to stresses, CUFs, flow- induced vibration, HELB locations, safety related valves and pumps, ISI program, and jet impingement and thrust forces. On the basis of this review described above, the NRC staff finds that the proposed MUR PU is acceptable.

3.6 Reactor Systems

3.6.1 Regulatory Evaluation

Early revisions of 10 CFR 50.46, and Appendix K to 10 CFR Part 50, required licensees to base their loss-of-coolant accident analysis on an assumed power level of at least 102 percent of the licensed thermal power level to account for power measurement uncertainty. The NRC later modified this requirement to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the preciously licensed power. The licensee proposed to use a Caldon LEFM CheckPlus system to decrease the uncertainty in the measurement of feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.36 percent.

The licensee developed its license amendment request consistent with the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

3.6.2 Technical Evaluation

The NRC staff reviewed the thermal-hydraulic aspects of the LEFM CheckPlus system installation, including its laboratory calibration, the effects of system changes such as transducer replacement, and the impact the system installation will have, if any, on the applicable plant safety analyses.

3.6.2.1 Feedwater Flow Measurement Device Installation

The LEFM CheckPlus Systems at North Anna Units 1 and 2 consist of measurement sections/spool pieces located in each 16-inch feedwater line. The feedwater lines, in sequence, contain the existing feedwater flow venturis, a 90-degree pipe bend, the LEFM CheckPlus spool piece, and the main feedwater check valves.

The licensee stated that the CheckPlus Ultrasonic Flow Meter (UFM) has to be installed in accordance with the requirements of Caldon Topical Reports ER-80P and ER-157P related to the LEFM Check and LEFM CheckPlus Systems. Each CheckPlus UFM is located in the 16-inch main feedwater line at the following distances from the 90-degree pipe bend:

Table 3.1 – UFM Downstream Centerline Distance from Pipe Bend

| NUMBER | FEEDWATER LINE | DOWNSTREAM CENTERLINE DISTANCE FROM PIPE BEND |
|---|---|---|
| UNIT 1 | A | 29 feet 3 inches |
| UNIT 1 | B | 14 feet 1 inch |
| UNIT 1 | C | 24 feet 0 inches |
| UNIT 2 | A | 18 feet 1 inch |
| UNIT 2 | B | 21 feet 7 inches |
| UNIT 2 | C | 28 feet 7 inches |

3.6.2.2 CheckPlus Non-Functionality

To operate above the presently licensed power of 2893 MWt, the licensee proposes that the CheckPlus cannot have been out-of-service for more than 48 hours and the plant exhibits steady-state conditions during the 48 hours. The feedwater UFM calorimetric is considered non-functional or out-of-service when either UFM system is not functional or the plant computer system is not functional.

Power during the 48 hours without an operational CheckPlus will be monitored using existing plant instrumentation, such as the feedwater flow venturis and RTDs. The licensee justifies this operation on the basis that alternate plant instrumentation exists to calculate calorimetric power, the flow rates detected by each venturi will be normalized to the UFM feedwater flow rate, and North Anna Units 1 and 2 have not had a history of venturi fouling.

If a power change occurs that lowers the power below 2893 MWt during the 48-hour window, the maximum permitted power level while the UFM is not functional is 2893 MWt. This approach is taken to simplify operator response and prevent the misdiagnosis of a failure mode. The actions are to be covered in the Technical Requirements Manual. The NRC finds that operation with an inoperable CheckPlus has been acceptably addressed.

3.6.2.3 Transducer Replacement

The effect of transducer replacement on the CheckPlus system uncertainties has been addressed in the letter dated March 26, 2009. The original uncertainty calculations included a transducer replacement uncertainty per LEFM CheckPlus meter for North Anna Unit 1 and for Unit 2. Since the uncertainty per the LEFM CheckPlus meter is incorporated in the original uncertainty calculation and no additional uncertainty terms need to be applied whenever a transducer is replaced, the NRC staff finds that transducer installation variability has been acceptably addressed.

3.6.2.4 CheckPlus Calibration

CheckPlus calibration was accomplished at Alden Laboratories. Licensee's application covers the test configuration. The NRC staff compared the test configuration to drawings and information in licensee's supplemental letter dated July 8, 2009, and noted the following:

An inconsistency was noted between the laboratory calibration setup and the piping run in that the model piping configuration was flipped, with reference to ground, to the actual piping configuration in both North Anna Unit 1 and Unit 2 of the plant. The effect of gravity on the flow characteristics in North Anna Unit 1 and Unit 2 is judged negligible for the flow rates used in the tests.

The NRC staff reviewed drawings and schematics provided in the supplemental letter dated July 8, 2009, and confirmed that, insofar as configuration is concerned, the laboratory configuration largely matched the in-situ configuration. The effect of gravity on the flow characteristics is judged as negligible because the Reynolds numbers in both the tests and the plant are well above 1,000,000 and gravitational effects are not significant.

The calibration studies were completed to determine the meter factor, a calibration coefficient, and also parametric tests to determine the meter sensitivity to upstream hydraulics. The tests were completed using previously applied procedures and laboratory measurement elements traceable to the National Institute of Standards and Technology. The NRC staff finds that the licensee's laboratory calibration was sufficiently fabricated to provide meaningful data based on the modeling of piping geometry of the UFM at North Anna Units 1 and 2.

3.6.2.5 Nuclear Steam Supply System

The Nuclear Steam Supply System (NSSS) design parameters provide the reactor coolant system (RCS) and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the NSSS analyses.

In all safety analyses except where noted in Table 3.2, the assumed initial power level was 102 percent of original licensed thermal power (OLTP). The sections following Table 3.2 discuss those safety analyses not performed assuming 102-percent OLTP as an initial condition, and explain why the NRC staff found those analyses acceptable.

3.6.2.6 Accident Analyses Bounded by Current Analysis of Record

In all analyses, the licensee referenced the current analysis of record, which used computer codes and methodologies previously approved by the NRC, for each accident and transient analysis. Unless noted, the analyzed core power level was 2951 MWt, 2.0 percent greater than the current licensed core power level of 2893 MWt and 0.37 percent greater than the MUR core power level of 2940 MWt. The staff reviewed and approved the licensee's transient and accident analyses at 2951 MWt conditions assumed for normal operations confirming that the acceptance criteria were still met under these conditions.

Results of the NRC staff's review are summarized in Table 3.2 below. In the sections that follow, discussion is provided for those analyses that were not performed at 102 percent of OLTP.

Table 3.2 – Accident and Transient Analyses

| Accident/Transient | Analyzed Core Power Level (-percent of 2893 MWt) | Analysis of Record Bounds MUR Uprate | NRC Staff Conclusion/Discussion |
|---|---|---|---|
| Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition | 0 | Yes | See Section 3.6.2.6.1 |
| Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power | 102 | Yes | Acceptable |
| Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) | 102 | Yes | Acceptable |
| Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error) | 101.7 | Yes | See Section 3.6.2.6.2 |
| Large Break Loss-of-Coolant Accident | 100 | Yes | See Section 3.6.2.6.3 |
| Steam Generator Tube Rupture | 102 | Yes | Acceptable |
| Waste Gas Decay Tank Rupture | N/A | Yes | See Section 3.6.2.6.4 |
| Major Secondary System Pipe Rupture (Main Steam Line Break) | 0 | Yes | See Section 3.6.2.6.5 |
| Major Secondary System Pipe Rupture (Main Feed Line Break) | 102 | Yes | Acceptable |
| Locked Reactor Coolant Pump Rotor | 102 | Yes | Acceptable |
| Fuel-Handling Accident | 102.2 | Yes | See Section 3.6.2.6.6 |
| Analyses to Determine EQ Parameters | 100.24 | Yes | See Section 3.6.2.6.7 |

3.6.2.6.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition

The uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition is evaluated at hot zero power conditions using NRC-approved methodologies. The analysis of record remains unaffected by the MUR power uprate and therefore, the analysis of record is acceptable for the PU.

3.6.2.6.2 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

The licensee evaluated the rod cluster control assembly misalignment at the dropped rod limit lines. The dropped rod limit lines determine the allowable radial peaking factor at the limiting point of the transient. This event is evaluated during each fuel cycle and has been evaluated for the MUR power uprate and remains bounding. Therefore, the rod cluster control assembly misalignment analysis is acceptable.

3.6.2.6.3 Large Break Loss-of-Coolant Accident

The licensee uses an NRC-approved methodology for a realistic large break loss-of-coolant accident. The realistic analysis uses a single power of 2893 MWt with an uncertainty range determined by the methodology. To evaluate for the MUR, the licensee completed a sensitivity study. The sensitivity study increased the nominal core power from 2893 MWt to 2942.2 MWt and reduced the one sigma standard deviation from 1.1 percent to 0.2 percent, corresponding to 31.8 MWt and 5.9 MWt, respectively, and a 0.3-percent power calorimetric uncertainty based on the new uncertainty associated with using the UFMs.

The licensee completed the sensitivity study. It was observed that there was no observed correlation between changes in the peak cladding temperature (PCT) and the changes in core power, the maximum PCT identified in the UFSAR have not changed, and the change in PCT is small for the change in core power parameters. The Unit 1 PCT increased by 2°F and the Unit 2 PCT increased by 20°F. Changes less than 30°F are considered under the threshold for low importance based on the AREVA Richland, Inc., *Realistic Large Break LOCA Methodology for Pressurized Water Reactors,* EMF-2103(P)(A).

The licensee completed an evaluation of 10 CFR 50.46 Acceptance Criteria in which it was determined that significant margin exists to the 2200°F PCT limit, 17-percent local oxidation limit, and the 1-percent core-wide oxidation limit. The licensee determined that the Unit 1 local oxidation increased by 0.14 percent and Unit 2 by 0.19 percent. The Unit 1 whole core oxidation did not change and the Unit 2 whole core oxidation increased by 0.01 percent. The licensee plans to apply a PCT penalty to the analysis of record to address the potential for cumulative effects of small changes. The Unit 1 penalty will be an increase of 2°F from 1925°F to 1927°F PCT and the Unit 2 penalty will be an increase of 20°F from 1919°F to 1939°F.

The NRC staff finds this exception acceptable on the basis that the new feedwater flow instrumentation has the demonstrated capability to perform within the bounds of the reduced uncertainty value and there is a penalty applied to the PCT analysis of record, such that the realistic LOCA analyses will continue to bound operation of the plant at the uprated power level of 2940 MWt.

3.6.2.6.4 Waste Gas Decay Tank Rupture

The licensee completed the waste gas decay tank rupture analysis as part of the original plant licensing basis. The analysis resulted in a 1.6 rem whole-body dose at the exclusion area boundary. The 10 CFR Part 100 acceptance criterion for waste gas decay tank rupture exclusion area boundary is 25 rem whole-body dose. There is sufficient margin in the conservatisms in the radiological atmospheric dispersion factor (X/Q), dose conversion factors and gap activities that were used in the analysis such that the MUR PU impact is bounded by the original analysis and is acceptable.

3.6.2.6.5 Major Secondary System Pipe Rupture (Main Steam Line Break)

The main steam line break accident analysis for departure from nucleate boiling ratio is limiting at 0-percent power. Therefore, the analysis of record remains bounding for the MUR PU and is acceptable.

3.6.2.6.6 Fuel-Handing Accident

The analysis of record fuel-handling accident is based on the alternate source term as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. This analysis was completed assuming a core inventory of 2958 MWt, or 102.2 percent of 2893 MWt with a single failed fuel assembly of 264 rods. The analysis of record bounds the MUR uprate, and is therefore, acceptable.

3.6.2.6.7 Analysis to Determine Environmental Qualification (EQ) Parameters

The analysis to determine the environmental qualification parameters for normal non-radiological plant operating conditions assumed within all environmental zones are unchanged for power uprate conditions. The licensee completed an evaluation to assess the potential increase in normal operation radiation and the power uprate operation would be expected to increase the core inventory of radioisotopes by the percentage increase in core power. Conservative factors such as the use of a dose for a given radiation zone designation that represents the maximum end of normal operation range and the limitation imposed by plan operations as a result of Technical Specification limits on reactor coolant system coolant activity are incorporated into current estimates and therefore the analysis of record is bounding.

The licensee evaluated the EQ parameters during accident conditions at post-accident radiation source terms at a core power of 2951 MWt and 18-month fuel cycle. The analysis is bounding for the MUR PU, and is therefore, acceptable.

3.6.3 Conclusion

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed license amendment request in support of implementation of a measurement uncertainty recapture. 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. Most of the current analyses of record are based on operation at 2951 MWt, which includes 2.0 percent measurement uncertainty. The proposed amendment is based on the use of a Caldon LEFM Check Plus system that would decrease the uncertainty in the feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.36 percent. In these cases, the proposed MUR rated thermal power of 2940 MWt is bounded by the current analyses of record.

## 3.7 Reactor Vessel and Internals Integrity

The NRC staff's review in the area of RPV integrity focuses on the impact of the proposed MUR PU on pressurized thermal shock (PTS) calculations, fluence evaluations, heatup and cooldown pressure-temperature (P-T) limit curves, low-temperature overpressure protection, upper shelf energy (USE), and RPV surveillance capsule withdrawal schedules. This review was conducted, consistent with the guidance contained in RIS 2002-03, to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR Part 50, Sections 50.60 and 50.61, Appendices G and H, following implementation of the proposed MUR PU.

### 3.7.1 RPV Material Surveillance Program

#### 3.7.1.1 Regulatory Evaluation

The RPV material surveillance program provides a means for determining and monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RPV. Appendix H of 10 CFR Part 50 provides the staff's requirements for the design and implementation of the RPV material surveillance program.

#### 3.7.1.2 Technical Evaluation

By letter dated July 1, 2004 (ADAMS Accession No. ML041950277), the licensee submitted changes to the Reactor Coolant System Pressure-Temperature (P-T) Limit Curves and Low Temperature Overpressure Protection System (LTOPS) setpoints, valid to the end of the 60-year license renewal period. The revised P-T limits and LTOPS setpoints are based on reactor vessel neutron fluence projections for the period of license renewal, corresponding to cumulative core burnups of 50.3 Effective Full Power Years (EFPY) for North Anna, Unit 1, and 52.3 EFPY for North Anna, Unit 2. Staff approved the revised P-T Limits and LTOP setpoints by letter dated July 8, 2005 (ADAMS Accession No. ML051860255). The peak RPV inner surface fluence values in these analyses were 5.90 E19 n/cm2 (E > 1MeV) and 5.91 E19 n/cm2 (E > 1MeV) for North Anna Units 1 and 2, respectively, at end of license (EOL). These values are conservative in comparison to those calculated for the MUR power uprate (5.14 E19 n/cm2 (E > 1MeV) and 5.25 E19 n/cm2 (E > 1MeV) for North Anna Units 1 and 2, respectively). The staff determined that the updated fluence values, which reflect the license renewal period, are bounding for this power uprate request.

By letter dated March 15, 2005 (ADAMS Accession No. ML050740524), the licensee submitted a proposed RPV surveillance capsule withdrawal schedule for the NRC staff's review and approval. Additional information was submitted on December 9, 2005 (ADAMS Accession No. ML0534701720). The staff found that the revised surveillance capsule withdrawal schedules for North Anna Units 1 and 2 satisfy the requirements of the guidance of the American Society for Testing and Materials (ASTM) E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" and NUREG-1802, "Generic Aging Lessons Learned (GALL) Report." In a Safety Evaluation (SE) dated February 17, 2006 (ADAMS Accession No. ML0604603931), NRC staff approved the revised surveillance capsule withdrawal schedule for the 60-year extended license period.

Since the fluence values for the MUR are bounded by the fluence values used to determined the license renewal EOL transition temperature shift values, the staff determined that the North Anna

Units 1 and 2 RPV surveillance programs will continue to meet the requirements of 10 CFR Part 50, Appendix H under the MUR PU condition.

3.7.2 P-T Limits and Use

3.7.2.1 Regulatory Evaluation

Appendix G or 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 on the upper-shelf energy (USE) values used for assessing the safety margins of the RPV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of the USE assessments covered the impact of the MUR PU on the neutron fluence values for the RPV beltline materials and the USE values for the RPV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of EFPY specified for the proposed MUR PU, considering neutron embrittlement effects.

3.7.2.2 Technical Evaluation

Regarding the topic of the RPV P-T limits, the licensee concluded in Attachment 5 Section IV.1.C.iii that:

> The current P-T Limit curves are licensed through 50.3 EFPY for Unit 1 and 52.3 EFPY for Unit 2. RTNDT calculations have been performed per Regulatory Guide 1.99, Revision 2 for NAPS, Unit 1 and Unit 2 reactor vessel beltline materials at the EOL neutron fluence values corresponding to 50.3 EFPY for Unit 1 and 52.3 EFPY for Unit 2. The most limiting 1/4T RTNDT value of 218.5°F bounds the EOL and MUR power uprate limiting material for both North Anna Units 1 and 2.

The NRC staff confirmed that the inner diameter (ID) neutron fluence for the proposed MUR PU is bounded by that in the SE dated July 8, 2005 (ADAMS Accession No. ML051860255), approving the current P-T limits and LTOP setpoints. Since the MUR PU neutron fluence is bounded by the current neutron fluence at 50.3 EFPY for North Anna Unit 1 and 52.3 EFPY for North Anna Unit 2, the MUR PU has no impact on the current P-T limit curves and LTOP setpoints. Therefore, the staff confirmed that the North Anna Units 1 and 2 RPV P-T limits and LTOP setpoints would continue to meet the requirements of 10 CFR Part 50, Appendix G under the MUR PU condition.

Regarding the topic of the RPV USE, the licensee concluded in Section IV.1.C.v of Attachment 5 to the submittal that the 1/4T USE values for North Anna Unit 1 and Unit 2 beltline materials meet the 50 ft-lb acceptance criteria of 10 CFR Part 50, Appendix G.

The NRC staff has evaluated the information provided by the licensee in the submittal, as well as information contained in NUREG-1766 (ADAMS Accession No. ML023090559), "Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2" and by letter dated December 13, 2005 (ADAMS Accession No. ML0534803582), in which the licensee submitted updated information for the reactor vessel integrity database, to reflect the license renewal period. Staff confirmed that the limiting 1/4T adjusted reference temperature (ART) value of 218.5°F for the North Anna Unit 2 lower shell forging heat number 990533/297355 is the bounding material for the extended period of operation for both units. Since the neutron fluence values for the USE under the MUR PU condition are bounded by the approved neutron fluence values for license renewal, the staff concludes that the USE would be bounded by the current analysis and that North Anna Units 1 and 2 RPV materials would continue to meet the USE criteria requirements of 10 CFR Part 50, Appendix G under the MUR PU condition. Therefore, the NRC staff finds the proposed MUR PU acceptable with respect to the P-T limits and USE.

### 3.7.3 PTS

#### 3.7.3.1 Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of PWR RPV beltline materials to failure during a PTS event to assure that adequate fracture toughness exists during reactor operation. The NRC staffs requirements, methods of, evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61. The NRC staffs review covered the PTS methodology and the calculations for the reference temperature for pressurized thermal shock (RTPTS) at the expiration of the license, considering neutron embrittlement effects.

#### 3.7.3.2 Technical Evaluation

Regarding the topic of PTS analyses for the North Anna Units 1 and 2 RPVs, the licensee provided the RTPTS values (190.9ºF and 227.7ºF) for the limiting beltline materials, lower shell forging 90400/292332 and lower shell forging 990533/297355, for North Anna Units 1 and 2, respectively. Staff has confirmed that these values are consistent with the RTPTS values reported in information contained in NUREG-1766 (ADAMS Accession No. ML023090559) and by letter dated December 13, 2005 (ADAMS Accession No. ML0534803582), using the more conservative neutron fluence values previously described. The screening criterion for these forgings is 270ºF. Since the neutron fluence values for the MUR PU are lower than those reported and approved in the license renewal analyses, the staff concluded that the RPV will remain within its limits for PTS after the MUR PU. The staff confirmed that the North Anna Units 1 and 2 RPV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61.

### 3.7.4 RPV Internals and Core Support Materials

#### 3.7.4.1 Regulatory Evaluation

The RPV internals and core supports include structures, systems and components (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 reactor coolant pressure boundary). The NRC's acceptance criteria for RPV internals and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RPV internals and core supports. Matrix 1 of NRC RS-001, Revision 0, "A Review Standard for Extended Power Uprates," provides references to the NRC's approval of the recommended guidelines for RPV internals in Topical Reports WCAP-14577, Revision 1-A, "A License Renewal Evaluation: Aging Management for Reactor Internals," March 2001, and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," March 2000.

3.7.4.2 Technical Evaluation

The licensee discussed the impact of the MUR PU on the structural integrity of the North Anna Units 1 and 2 RPV internal components in Attachment 5 Section IV.1.A.ii of the submittal. The licensee concluded that the temperature changes due to the MUR PU are minimal and, therefore, the existing loads remain valid and the stresses and fatigue values also remain valid.

The RPV internals of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC);
- loss of fracture toughness properties induced by radiation exposure for all stainless steel grades, or the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;
- stress relaxation in bolted, fastened, keyed or pinned RPV internal components induced by irradiation exposure and/or exposure to elevated temperatures; and
- void swelling (induced by radiation exposure).

Table Matrix 1 of NRC RS-001, Revision 0 provides the staff's basis for evaluating the potential for extended PU to induce these aging effects. In Note 1 to Table Matrix 1, the staff stated that guidance on the neutron irradiation-related threshold for IASCC for PWR RPV internals are given in BAW-2248-A and WCAP-14577, Revision 1-A. This Table Matrix 1 note further stated that for thermal and neutron embrittlement of CASS, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs that investigate degradation effects and determine appropriate management programs.

In NUREG-1766 (ADAMS Accession No. ML023090559), the licensee stated that it would remain active in industry groups to stay aware of new industry developments regarding such issues as void swelling, neutron embrittlement of baffle and barrel bolting, and thermal embrittlement of CASS components. Since neutron fluence values are bounded by the values in NUREG-1766 and changes in operating temperatures are minimal, staff concludes that the MUR PU is not expected to have any significant impact on irradiation-related aging degradation of the RPV internals.

3.7.5 Conclusion

The NRC staff has reviewed and evaluated the impact that the MUR PU conditions will have on the structural integrity assessments for the RPV and RPV internals. It is determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) RPV surveillance program; (2) RPV USE assessment; (3) P-T limits; (4) PTS assessment; and (5) RPV internals. Therefore, the NRC staff finds that sufficient information has been provided to support the requested MUR PU.

## 3.8 Electrical Systems

### 3.8.1 Regulatory Evaluation

The licensee developed the license amendment request (LAR) consistent with the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture (MUR) Uprate Applications."

The regulatory requirements which the staff applied in its review of the application include: General Design Criterion (GDC) 17, "Electric power systems," of Title 10 to the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety.

Section 50.63 requires that all nuclear plants have the capability to withstand a loss of all alternating current (AC) power (station blackout (SBO)) for an established period of time, and to recover therefrom.

Section 50.49, "Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

### 3.8.2 Technical Evaluation

#### 3.8.2.1 AC Distribution System

The AC Distribution System is the source of power for the nonsafety-related buses and for the safety-related emergency buses. It consists of the 4.16 kiloVolt (kV), 480 V, and 120 V systems. The onsite power distribution loads were reviewed under normal and emergency operating scenarios. At the uprated power level, the loads are expected to operate within their design ratings. The licensee stated that the MUR PU will only affect the 4.16 kV buses while the 480 V, 120 V and 125 VDC buses will not see load increases. The 4.16 kV loads that will be affected are the main feedwater pump, condensate pump, low pressure heater drain pump, high pressure heater drain pump, and reactor coolant pump. The licensee further stated that the brake horsepower values for these pumps will not exceed their motor nameplate ratings. Based on this information, the staff finds that the existing AC distribution system will be able to support the loading for uprated conditions.

#### 3.8.2.2 Power Block Equipment (Generator, Exciter, Transformers, Iso-phase bus duct, Generator circuit breaker)

For Unit 1, the generator is rated at 1105 megavolt ampere (MVA) and is operated with restrictions not to exceed 475 megavolt ampere reactive (MVARs) out or 390 MVARs in. Currently, the generator output is 965 megawatt electric (MWe). At uprated conditions, the output will be 980.5 MWe with 472.9 MVAR out and 390 MVAR in. The increase in electrical output (15.5 MWe) remains bounded by the design ratings of the generator. The licensee stated in the LAR, that the new operating point of the generator is below the main generator maximum capability. Based on this information, the staff finds that the Unit 1 generator is capable of operation at uprated conditions.

For Unit 2, the generator is rated at 1200 MVA and is operated with restrictions not to exceed 444 MVARs out or 210 MVARs in. Currently, the generator output is 965 MWe. At uprated conditions, the output will be 972.9 MWe with 444 MVAR out and 210 MVAR in. The increase in electrical output (7.9 MWe) remains bounded by the design ratings of the generator. The licensee stated in the LAR, that the new operating point of the generator is below the main generator maximum capability. Based on this information, the staff finds that the Unit 2 generator is capable of operation at uprated conditions.

The iso-phase bus is rated for 30,500 A. The MUR PU will increase the current on the iso-phase bus to 30,072 A for Unit 1 and 29,862 amperes for Unit 2. Therefore, the staff finds that the iso-phase bus is capable of operation at uprated conditions since the increase from the MUR PU remains below the iso-phase bus rating.

The main generator circuit breaker is installed only on Unit 1, which allows the normal station service buses to be supplied from the normal station service transformers when the main generator is offline. The main generator circuit breaker has a continuous current rating of 36,000 amperes and the licensee stated that the current rating has margin to the maximum generator output at MUR PU conditions.

Each main generator feeds electric power through a 22 kV isolated phase bus to a bank of three single-phase transformers for each unit, stepping the generator voltage of 22 kV up to the transmission voltage of 500 kV. These main (step-up) transformers are rated for 1200 MVA. The licensee stated, in the July 8, 2009, letter, that the uprated loadings of the main transformers are 1088.6 MVA minus the station service transformer loadings, which is below the rating of the main transformers (1200 MVA). Therefore, the staff finds that the main (step-up) transformers are capable of operation at uprated conditions.

Station service transformers, connected to the 22 kV isolated phase bus from each main generator, normally supply power to the auxiliaries of each unit by stepping down the 22 kV to 4.16 kV. During normal operating conditions the station service transformers power the 4.16 kV switchgear, 480 V load centers and motor control centers. The ratings of the plant service transformers are 22.4 MVA, as stated in the licensee's July 08, 2009, letter. Furthermore, at uprated conditions, the loadings for any of the station service transformers are within the design rating of the plant service transformer of 22.4 MVA. Therefore, the staff finds that the station service transformers are capable of operation at uprated conditions.

Three 3-phase 34.5/4.16-kV reserve station service transformers receive 34.5 kV supply from two 500/36.5 kV transformers and one 230/34.5-kV transformer, and supply reserve station service power, for start-up and emergency use. In the July 8, 2009, letter, the licensee stated that the uprated loadings of the reserve station service transformers are 19.0 MVA, 29.7 MVA, and 30.5 MVA for reserve station service transformers A, B, C, respectively, which is below the 33.6 MVA design rating of the reserve station transformers. Therefore, the staff finds that the reserve station service transformers are capable of operation at uprated conditions.

3.8.2.3 DC System

The 125V DC power source in each unit consists of four independent batteries and six battery chargers with two chargers as spares. The 125V DC distribution system for each unit consists of four main distribution cabinets for plant operation and DC loads essential for unit safety. Each standby diesel generator has an independent 125V DC power system.

The licensee stated that the MUR PU does not affect any DC powered indication, control, or protection equipment. The staff reviewed the LAR and FSAR and confirmed that the PU does not impact DC system loads. Therefore, the staff finds that the analyses for the DC system bound MUR PU conditions.

3.8.2.4 Emergency Diesel Generators (EDG)

The EDG system provides a safety-related source of AC power to sequentially energize and restart loads necessary to shut down the reactor safely, to maintain the reactor in a safe shutdown condition, and operate all auxiliaries necessary for safety. The EDG system is capable of performing this function during a loss of offsite power. Each EDG (two per unit) is dedicated to one of the 4160 V ESF buses (Class IE), which supply power to critical loads required during abnormal operational transients and accidents.

The licensee stated in the LAR that there are no increases to the emergency buses loads supported by the EDGs, and thus, the existing accident analyses remain bounding. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR PU conditions

Based on the above, the staff, after reviewing the LAR and UFSAR, finds that the PU does not impact EDG system loads. Therefore, the staff finds that the analyses for the EDG system bound MUR PU conditions, and the onsite power system will continue to meet the requirements of GDC 17.

3.8.2.5 Switchyard

The switchyard equipment and associated components are classified as nonsafety-related. The switchyard serves three 500 kV lines and one 230 kV line. The plant is connected to the switchyard by two 500 kV transmission lines, three 34.5 kV underground lines and two 34.5 kV overhead that supply power to the three reserve station service transformers. The primary function of the switchyard and distribution system is to connect the station electrical system to the transmission grid.

The small increase in plant output does not significantly impact the switchyard equipment, since there is sufficient margin. Therefore, the staff finds that the analyses for the switchyard system for North Anna Units 1 and 2 bound the MUR PU conditions.

3.8.2.6 Grid Stability

The licensee stated in its LAR and the subsequent supplemental letters that grid stability studies were conducted by Pennsylvania, New Jersey, and Maryland Interconnection (PJM) and that the evaluation included load flow studies, short circuit studies, and stability analysis. The licensee provided this information by referring the PJM study posted on public website. The evaluation assessed the system at the summer 2012 conditions. The licensee stated that the local generation study identified no transmission deficiencies and indicated no decrement to system First Contingency Incremental Transfer Capability. The licensee stated that in the summary section of the PJM impact studies, the maximum facility output is listed as 945 MWe for Unit 1 and 938 MWe for Unit 2. However, the output of the generators, as a result of the power uprate, will be 980.5 MWe for Unit 1 and 972.9 for Unit 2.

The staff requested additional information on the grid stability study, specifically asking the licensee to explain the results of the study and how the results of the study are valid, given the generator output. In its July 8, 2009, response, the licensee stated that for Unit 1, 980.5 MWe was the gross generator output and by subtracting off the 48 MWe of internal loads, the net maximum facility output would be 932.5 MWe, which is below the 945 MWe used in the PJM study. Similarly, for Unit 2, subtracting 48 MWe of internal loads from the gross generator output of 972.9 MWe gives a net maximum facility output of 924.9 MWe, which is below the 938 MWe used in the PJM study. If a unit trips, the grid would experience a transient comparable to the loss of the gross output and internal loads would still remain, as needed to be powered from the grid. Hence, the grid stability analyses need to demonstrate that the grid can adequately handle this disturbance. In its July 16, 2009, letter, the licensee stated that the system impact studies considered the associated gross generator outputs of 1075 MWe for Unit 1 and 1068 MWe for Unit 2. These studies addressed an additional 85 MWe per unit, due to the PU (20 MWe) and also increased generation from turbine replacements (65 MWe).

In its July 24, 2009, supplemental letter, the licensee stated that a final impact study was performed by PJM Interconnection to determine the impacts of the expected increase in MWe. The study concluded that there were no transmission deficiencies, no decrement to First Contingency Incremental Transfer Capability between utilities, and no transient stability issues. Thus, the impact study bounds the increase in MWe of each North Anna Unit from the PU.

The staff requested additional information regarding how the studies ensured adequate post-trip voltages. Specifically, when a unit trips, the grid would lose generation but still needs to power the internal or house loads. The licensee stated in its July 24, 2009, letter that the gross generator output was used in the dynamic stability studies and that PJM evaluated external unit trip disturbances such that the station loads are maintained as loads on the grid. Based on this information, the staff finds that the grid stability studies ensure adequate post-trip voltages since the gross output was utilized in the studies and the generator and internal loads are modeled separately.

The staff reviewed the grid stability study, and finds that the North Anna Units 1 and 2 MUR PU allows for continued stable and reliable grid operation.

3.8.2.7 Station Blackout (SBO)

Section 50.63 requires that each light water cooled nuclear power plant be able to withstand and recover from a loss of all AC power, referred to as an SBO.

North Anna's SBO coping duration is 4 hours. This is based on the licensee's evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 and Regulatory Guide 1.155. The offsite power design characteristics include the expected frequency of a grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power.

The licensee stated that the evaluation for SBO included the adequacy of alternate ac power source, emergency condensate storage tank inventory, class 1E battery capacity, ventilation, compressed air, and containment isolation. The alternate AC power source has sufficient capacity to operate systems necessary for coping with an SBO event for the required coping period. The licensee stated that the emergency condensate storage tank inventory is adequate for decay heat removal following an SBO event at uprated conditions. The proposed MUR has no affect on North Anna's station battery capacity as the MUR does not increase loads. The ventilation for areas containing SBO equipment, the operation of air-operated or containment isolation valves and the ability to restore compressed air is unaffected by the MUR PU. Based on this information, the staff finds that the MUR PU will have no impact on North Anna's SBO coping duration. Furthermore, the SBO analyses used 2951 MWt (102-percent of 2893 MWt) remain bounding. Therefore, the staff finds that North Anna Units 1 and 2 will continue to meet the requirements of 10 CFR 50.63 under PU conditions.

3.8.2.8 Environmental Qualification (EQ) Program

In the LAR, the licensee stated that the temperature, pressure, and humidity remain unchanged within all environmental zones as a result of the MUR PU, for both normal operation and accident conditions. However, the PU will increase the radiation source term used in the environmental qualification (EQ) program. The licensee stated in its LAR that the increase in the radiation source term will not affect the currently estimated normal operation doses used for North Anna's EQ program since conservative factors were used. Specifically, the licensee used the maximum end of the normal operating range for a radiation zone, and limitations on coolant activity imposed by Technical Specifications. Therefore, the licensee stated that with the exception of the reactor vessel excore neutron detectors the MUR PU is bounded by the EQ program for normal operation.

The licensee stated in its LAR that the normal operation radiation dose levels increase as a result of the MUR PU for the reactor vessel excore neutron detectors and the qualified life of the excore detectors may be impacted. Furthermore, the licensee concludes that the preliminary results indicate no impact on radiation dose margin or qualified life of the excore detectors. Based on the fact that the there might be a decreased EQ in-service life of the excore detectors due to the increase in radiation dose, the licensee provided a Regulatory Commitment to incorporate changes in the qualified lifetime of this equipment into the EQ program documentation prior to operating above the present reactor thermal power. The licensee shall determine the environmental qualification service life of the excore detectors and incorporate changes in the qualified lifetime of this equipment into environmental qualification program documentation, prior to operating above the current maximum operating level of 2893 MWt, as described in Virginia Electric and Power Company's letters dated March 26, 2009, July 8, 2009, and July 24, 2009.

NRC Safety Evaluation dated March 22, 2005 (ADAMS Accession No. ML050840156), determined that hydrogen recombiners are not required to be included in the Technical Specifications and the requirement for these components has been deleted from 10 CFR 50.44. As a result, the licensee is removing the hydrogen recombiners and hydrogen monitoring equipment from the EQ program. The staff finds this acceptable.

Based on this information, the staff finds that the current EQ qualifications remain bounding with exception of excore detectors for the MUR PU. Therefore, the staff finds that with the exception of the excore detectors the MUR PU will have no impact on North Anna's EQ Program and continues to meet the requirements of 10 CFR 50.49.

### 3.8.3 Conclusion-Overall Electrical Systems

Based on the technical evaluation provided above, the NRC staff finds that the licensee will continue to meet GDC 17, 10 CFR 50.63, and 10 CFR 50.49. As stated above, this safety evaluation includes completion of actions to incorporate changes in the qualified lifetime of reactor vessel excore neutron detectors into the equipment qualification program. Therefore, the staff finds the MUR PU acceptable.

## 3.9 Instrumentation & Controls

### 3.9.1 Regulatory Evaluation

Topical Report ER-80P and its supplement, Topical Report ER-157P, describe the Caldon LEFM CheckPlus System for the measurement of feedwater flow and provide a basis for the proposed 1.6 percent MUR uprate of the licensed reactor thermal power. The NRC staff also considered guidance of Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," in its review of the licensee's submittals for the proposed power uprate request.

### 3.9.2 LEFM Technology and Measurement

Both the Caldon LEFM Check and LEFM CheckPlus systems use 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 fluid 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, and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Both systems use 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 feedwater 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 in a plant-specific piping model at a calibration laboratory.

The Caldon LEFM Check system consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The Caldon LEFM CheckPlus system uses 16 transducers, 8 each in two orthogonal planes of the spool piece. In the Caldon 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. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The NRC staff's review in the area of instrumentation and control covers the proposed plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique, in accordance with the guidelines (A through H) provided in Section I of Attachment 1 to RIS 2002-03. The staff conducted its review to confirm that (1) the licensee's implementation of the proposed feedwater flow measurement device is consistent with the staff-approved Topical Reports ER-80P and ER-157P and (2) the licensee adequately addressed the four additional requirements listed in the staff's SE. The staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.36 percent correctly accounted for all uncertainties associated with power level instrumentation errors, and (2) the uncertainty calculations met the relevant requirements of Appendix K to 10 CFR Part 50, as described in Section 2 of this SE.

The licensee provided the following information regarding the Caldon LEFM CheckPlus System feedwater flow measurement technique and its implementation at North Anna Units 1 and 2. The Caldon LEFM CheckPlus System consists of an electronic cabinet installed in the cable spreading room and two spool piece measurement sections installed in each of the three main feedwater flow lines between the existing feedwater venturi flow meters and the main feedwater check valves. Each measurement section consists of 16 ultrasonic, multipath, transit time transducers, 1 dual resistance temperature detector (RTD), and 2 pressure transmitters. The electronic cabinet controls magnitude and sequences transducer operations; makes time measurements; and calculates volumetric flow, temperature, and mass flow. The system software measures velocities at precise locations. The feedwater mass flow rate and temperature are transmitted to the plant process computer for use in calorimetric measurement of reactor thermal output. In the event of system failure, including loss of power and electronic cabinet high temperature, the control room operators are alerted via an annunciator window for "Feedwater Ultrasonic Flow Meter Failure."

The UFM values for feedwater mass flow and temperature will be directly substituted for the existing venturi-based flow and RTD temperature inputs used in the plant calorimetric measurement calculations. The existing venturi-based feedwater flow and RTD temperature will continue to be used for other plant functions and may be used for plant calorimetric calculations in the event of a UFM failure.

3.9.3 LAR Compliance to RIS 2002-03, Attachment 1, Section I, Guidance A through H

Items A through C

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03, respectively, guide licensees in identifying the approved topical reports, providing references to the NRC's approval of the measurement technique, and discussing the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the feedwater flow measurement technique. In this LAR, the licensee identified Topical Reports ER-80P and ER-157P as applicable to the Caldon LEFM CheckPlus System. The licensee also referenced NRC SERs for Topical Reports ER-80P, dated March 8, 1999, and ER-157P, dated December 20, 2001. The licensee also cited an NRC SER dated July 5, 2006, which reexamined the performance of the Caldon systems and confirmed the validity of the previously referenced Caldon topical reports. The licensee stated in its submittal that the Caldon LEFM CheckPlus System is to be permanently installed in North Anna Units 1 and 2, per the requirements of Topical Reports ER-80P and ER-157P. The Caldon LEFM CheckPlus System was installed in North Anna Unit 2 during the fall of 2008. The Caldon LEFM CheckPlus System was installed in North Anna Unit 1 during the spring of 2009.

Based on its review of the licensee's submittals as reflected in the above discussion, the staff finds that the licensee has sufficiently addressed the plant-specific implementation of the Caldon LEFM CheckPlus System using proper topical report guidelines. Therefore, the licensee's description of the feedwater flow measurement technique and implementation of the PU using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03.

Item D

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees in addressing four criteria when implementing the feedwater flow measurement uncertainty technique. The staff SERs on Topical Reports ER-80P and ER-157P both include these four plant-specific criteria to be addressed by a licensee referencing these topical reports for PU. The licensee's submittal addresses each of the four criteria as follows:

Criterion 1

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

Licensee's implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM CheckPlus System. Work will be performed at North Anna Units 1 and 2 by site instrumentation and control personnel qualified by the North Anna Units 1 and 2 Instrumentation and Control Training Program.

In addition, the licensee states that a preventive maintenance program for the Caldon system will be developed using the vendor's maintenance and troubleshooting manual. Preventive maintenance activities for the Caldon LEFM CheckPlus System at North Anna Units 1 and 2 will include the following:

- general terminal and cleanliness inspection
- power supply inspection
- central processing unit inspection
- acoustic processor unit checks
- alarm relay checks
- watchdog timer checks to ensure that the software is running
- communication checks
- transducer checks
- calibration checks on each feedwater pressure transmitter

Based on the review of the licensee submittals, the staff concludes that the licensee adequately addressed Criterion 1.

Criterion 2

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

Licensee states that the Caldon LEFM CheckPlus System was only recently installed in North Anna Unit 2 and has only a few months of operating experience at the plant (as of the date of the LAR submittal). The licensee indicated that no maintenance activities have taken place during that time and that the system is operating as designed.

The Caldon LEFM CheckPlus System for North Anna Unit 1 was still scheduled for installation as of the date of LAR submittal; thus, no plant-specific maintenance experience exists. Given the very short time since the installation of the system in Unit 2, the staff finds the licensee's response adequate to address Criterion 2.

Criterion 3

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

The licensee uses a core thermal power uncertainty calculation consistent with ISA-RP67.04.02-2000 and the referenced Caldon topical reports, ER-80P and ER-157P. The combination of errors within instrument loops is performed using the licensee Technical Report EE-0116. The NRC recently reviewed EE-0116 and the licensee's associated setpoint methodology in the review of a separate license amendment (ADAMS Accession No. ML072681096). The NRC's SER for that review (ADAMS Accession No. ML082250013) concluded that the "licensee setpoint methodology meets the guidance provided in RIS 2006-17" and that the instrument setpoint calculations in EE-0116 "are calculated based on the acceptable ISA methodology."

Based on the foregoing, the NRC staff concludes that the licensee adequately addressed Criterion 3.

Criterion 4

> For plant installation where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors are not representative of the plant-specific installation), licensees should provide additional justification for its use. The justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, licensees

should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Licensee stated that the Caldon LEFM CheckPlus System was calibrated using a site-specific piping configuration at Alden Research Laboratories. In the LAR, the licensee submitted ER-689, Revision 2, "LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for North Anna Unit 1," and ER-675, Revision 2, "LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for North Anna Unit 2," both dated January 2009. According to ER-689, the meter factor uncertainty for Unit 1 is 0.20 percent. ER-675 gives the meter factor uncertainty for Unit 2 as 0.19 percent.

The licensee states that acceptance of the final site-specific uncertainty analyses will occur following completion of the commissioning process. The commissioning process will verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments. Final commissioning is expected to be complete by March 2010.

Based on the foregoing, the NRC staff concludes that the licensee adequately addressed Criterion 4. In addition, the licensee committed to confirming that the in situ test data are bounded by the calibration test data after final commissioning of the Caldon LEFM CheckPlus System.

Item E

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

To address Item E of RIS 2002-03, the licensee provided Cameron engineering reports ER-646, Revision 2, and ER-637, Revision 3. While reviewing the calculation in response to a request for additional information, the licensee discovered a minor error in the value used for steam moisture uncertainty in both documents. The licensee's July 8, 2009, supplemental submittal provided change pages for the LAR, as well as ER-646 and ER-637 (now Revision 3 and Revision 4, respectively, both dated June 2009).

The NRC staff reviewed the calculations and determined that the licensee identified all the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties (including those discussed in Item D.4 above), and calculated the overall thermal power uncertainty.

The licensee's calculations arithmetically summed uncertainties for parameters that are not statistically independent and statistically combined with other parameters. The licensee combined random uncertainties using the square root sum of squares approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of the Caldon LEFM CheckPlus System uncertainty, as described in the referenced topical reports and is consistent with the guidelines in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation."

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty. Therefore, the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees in providing information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.
In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03 related to all instruments that affect the power calorimetric as follows:

(1) Maintaining Calibration

The licensee states that calibration will be performed using procedures based on the appropriate LEFM CheckPlus technical manuals. Other calorimetric process instrumentation and computer points are maintained and calibrated using approved procedures. The response to Item D.1 above addresses the preventive maintenance program.

(2) Controlling Hardware and Software Configuration

The Caldon LEFM CheckPlus System is designed and manufactured in accordance with the vendor's quality assurance program, which meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The licensee committed to maintaining, after installation, the software and hardware configuration by using existing procedures and processes, which include verification and validation of software configuration changes. The UFM hardware and the calorimetric process instrumentation will be maintained per the North Anna Units 1 and 2 configuration control processes.

(3) Performing Corrective Actions

North Anna Units 1 and 2 personnel will monitor plant instrumentation that affects the power calorimetric input, including UFM inputs. Any problems detected are handled according to the North Anna Units 1 and 2 corrective action process.

(4) Reporting Deficiencies to the Manufacturer

The licensee states that any conditions found to be adverse to the quality of the LEFM CheckPlus System will be documented and reported to the vendor, as needed, to support corrective action.

(5) Receiving and Addressing Manufacturer Deficiency Reports

The licensee states that it has existing processes to address the receipt of manufacturer's deficiency reports. Any such deficiencies will be documented and controlled by existing processes.

The NRC staff's review of the above statements found that the licensee addressed the calibration and maintenance aspects of the Caldon LEFM CheckPlus System and all other instruments affecting the power calorimetric. Thus, the licensee meets the guidance in Item F of Section I of Attachment 1 to RIS 2002-03.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded.

In the LAR, the licensee proposed a 48-hour AOT for operating above the current licensed thermal power limit (2,893 MWt) if the UFM becomes nonoperational. The licensee specifically noted that any failure of the UFM, including failures of a single path or plane, will be treated as a complete failure of the UFM system and thus start the plant's allowable 48 hours to remain above 2,893 MWt. In addition, the ability of the plant to stay above 2,893 MWt is contingent upon its ability to maintain steady-state conditions. If the plant should reduce power below 2,893 MWt during the AOT, the plant will not be permitted to return above that value until the UFM function is fully restored.

During the 48-hour AOT in the event of a UFM failure, the plant would use alternate plant instruments (i.e., the existing feedwater venturis and RTDs) for the feedwater flow rate calculation. The venturis are regularly normalized to the UFM measurements, so their measurements should be equivalent to the UFM over the 48-hour AOT. Venturi nozzle fouling and transmitter drift were considered as potential sources of error within the AOT window. Recent inspections at both North Anna units showed no evidence of venturi fouling since plant startup; this makes fouling within a 48-hour period highly unlikely. In the July 8, 2009, submittal (ADAMS Accession Nos. ML091910204 and ML091910205), the licensee noted that the transmitters had recently been replaced in both units, so extensive history on their drift does not exist to justify the 48-hour AOT. However, the AOT time proposed is consistent with other measurement uncertainty recapture amendments that the NRC has approved. In addition, the licensee has made a commitment to confirm that the normalization factors applied to the venturis are negligible over a 48-hour period and are an acceptable surrogate for the LEFM measurements before their use above 2,893 MWt. A loss of the plant computer is treated as a loss of the UFM and the ability to obtain corrected calorimetric power using alternate instrumentation. Operation at the uprated power may continue until the time, not to exceed 24 hours, when the next required nuclear instrument heat balance is to be performed.

The NRC staff reviewed the licensee's submittals and found that the licensee provided sufficient justification for the proposed AOT and the proposed actions to reduce power level if the AOT is exceeded. Therefore, the licensee has followed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03.

3.9.4 Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations and determined that the licensee's proposed amendment is consistent with the staff's approved Topical Reports ER-80P and its supplement ER-157P. The staff has also determined that the licensee adequately accounted for all instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of

10 CFR Part 50, Appendix K as described in Section 2 of this SE. The licensee has committed to the following actions:

- Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments (per Item D.4 above).

- Confirm that the normalization factors applied to the venturis are negligible over a 48-hour period and are an acceptable surrogate for the LEFM measurements before their use above 2,893 MWt (per Items G and H above).

3.10 Plant Systems

3.10.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR PU on the NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and ESF HVAC systems. The staff's review is based on the guidance in SRP Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems. This evaluation is reflected in Attachment 5 of the licensee's application dated March 26, 2009.

The staff reviewed the licensee's license amendment request for compliance with the following regulations. GDC 16, Containment design, requires that the containment shall provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment. GDC 19 Control Room, requires that the control room must provide the operators with the capability to operate the nuclear power units safely under normal conditions and maintain the reactor in a safe condition under accident conditions including a LOCA. GDC 38, Containment heat removal, requires that the containment heat removal systems are capable of rapidly reducing the containment temperature and pressure following a LOCA and maintaining them at an acceptably low level. GDC 50, Containment design basis, requires that the containment accommodate the pressure and temperature conditions resulting from a LOCA without exceeding the design leakage rate. GDC 60, Control of releases of radioactive materials to the environment, requires that the nuclear power unit have means to control the release of radioactive materials in gaseous and liquid effluents during normal operation and anticipated operational occurrences.

3.10.2 Technical Evaluation

3.10.2.1 Containment Systems

The staff reviewed the following areas of containment design and analysis for the proposed North Anna Units 1 and 2 MUR PU: long-term LOCA containment response analyses, short-term LOCA containment response analyses, containment response to a main steam line break inside containment, impact of the MUR PU on containment isolation, and the impact of the MUR PU on combustible gas control.

The long-term mass and energy release analyses for North Anna Units 1 and 2 assumed 102.6 percent of 2893 MWt. Thus, these analyses remain bounding for the MUR PU. The short-term LOCA mass and energy releases (containment subcompartment response) analyses are affected by the fact that the cold leg temperatures are less than the current analyses.

Therefore, the mass flux into the subcompartments would increase for a cold leg break. However, calculations performed by the licensee show that this increase is insignificant. Therefore, the MUR PU is acceptable with respect to the short-term LOCA analyses.

The licensee states that the analyses of the main steam line break accident inside containment are conservative and bounding for the MUR PU. This is the expected result and the staff concurs.

The licensee states that the PU does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is independent of the reactor power level except for the possibility of overpressurization between redundant closed containment isolation valves addressed in Generic Letter 96-06. Since the predicted containment accident temperatures following the design-basis LOCA and main steam line break accident inside containment are bounded by those of the current licensing basis, this issue is acceptably addressed for the MUR PU.

The NRC issued a revision to 10 CFR 50.44, "Combustible gas control for nuclear power reactors," on September 11, 2003. The licensee requested a license amendment consistent with the changes to this rule. The NRC approved this amendment on Elimination of Requirements for Hydrogen Recombiners and Hydrogen Monitors Using the Consolidated Line Item Improvement Process, by letter dated March 22, 2005. The license amendment permitted the licensee to remove the hydrogen recombiners. The licensee states in its March 26, 2009, MUR PU request that these components are being removed.

#### 3.10.2.2 Heating Ventilation and Air Conditioning (HVAC)

The staff reviewed the impact of the MUR PU on the containment ventilation system, the control room ventilation system, the engineered safety features (ESF) ventilation system and the fuel-handling area ventilation system.

The licensee states that the additional containment heat load due to the MUR PU is insignificant. The staff concurs. Therefore, operation of the containment ventilation systems at MUR PU conditions is acceptable.

The licensee evaluated the control room heat loads (electrical, lighting, personnel) at MUR PU conditions and concluded that the main control room and computer room, relay rooms, emergency switchgear room normal and emergency ventilation systems and chilled water systems are not impacted by the MUR PU because the heat loads in these areas do not increase. The licensee evaluated the current limiting ESF heat loads and found no significant increase. This is expected and the staff finds operation of the ESF ventilation system at MUR conditions to be acceptable.

The licensee states that the spent fuel pool cooling equipment loads are not impacted by the MUR PU. The staff concurs.

3.10.2.3 Radioactive Waste Systems

The waste processing systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents. These waste systems and their various subsystems and components were evaluated for the PU. Systems are common to both units and are sized to treat the radioactive solid waste produced during simultaneous operation of both units. System functions and the waste processed volume are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The radioactive waste systems are bounded by the existing system design parameters and are acceptable at PU conditions. The NRC staff reviewed the licensee's assessment. The staff does not expect a 1.6-percent increase in power to result in a significant change to the operation of the radioactive waste systems. Therefore, based on the licensee's assessment, the staff finds that the radioactive waste systems will function adequately for the MUR PU.

3.10.2.4 Safety-Related Cooling Water Systems

The safety-related cooling water systems include the service water (SW) system, and the component cooling water (CCW) system, Ultimate Heat Sink, and Residual Heat Removal System.

The Component Cooling Water (CCW) system is a closed loop piping system shared between Units 1 and 2, and rejects heat to the SW system. There are four CCW pumps and four CCW heat exchangers, which can be cross-connected to share loads between the two units. Normally, two heat exchangers and two pumps (one per unit) are required to support the normal heat loads of both units. The CCW system is designed to provide the cooling requirements for normal plant operation, plant cooldown, spent fuel pool cooling and design basis accident cooldown of one unit. The licensee has evaluated the CCW system to confirm that the heat removal capabilities are sufficient to satisfy the MUR PU heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. The licensee stated that the analysis confirms that at MUR uprated conditions, normal plant operation and required cooldown time continue to be met.

The Service Water (SW) system consists of four SW pumps. Each pump takes suction from the SW reservoir, with two pumps required for the normal cooling requirements of both units. The SW system is designed to support a LOCA in one unit, while placing the non-accident unit in a cold shutdown condition in conjunction with a LOOP on both units. During an accident condition, three SW pumps are necessary to provide adequate heat removal for both units. Each component cooled by the SW system was evaluated to confirm that the existing flow rate is sufficient to satisfy the PU heat removal requirements during normal, shutdown, and accident conditions. The licensee's evaluations determined that the existing SW flows will continue to support the heat removal requirements at uprate conditions. The licensee states that the SW system and component design parameters remain bounding for PU operation. No system modifications are required to support the PU.

The ultimate heat sink is common to both units. The SW system inlet temperature for normal, shutdown, and accident conditions is bounded for the PU. The licensee states the ultimate heat sink is capable of cooling the SW system to prevent SW temperature from exceeding the inlet temperature limits during operating conditions, no system modifications are required to support the PU. Therefore, the ultimate heat sink is acceptable for operation at PU conditions.

The Residual Heat Removal (RHR) system cooldown performance was analyzed under MUR uprate conditions. The normal two train cooldown, single train cooldown, one RHR pump with two RHR heat exchanger cooldown and accident case cooldown were analyzed. The licensee stated that the analysis showed that each of these cases met the cooldown time requirements.

The NRC staff reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the licensee's determination that the existing analyses for these systems were evaluated for 102 percent RTP, the staff finds there is reasonable assurance that the systems are acceptable for the MUR PU.

3.10.2.5 Spent Fuel Pool Cooling (SFPC) and Purification System

The licensee stated there are no changes to the SFPC system limiting temperatures, pressures or flow rates as a result of the power uprate. Uprate conditions are bounded by the existing system design conditions. The principal function of the SFPC system is to provide storage and cooling of the spent fuel. The primary impact of a PU would be to the decay heat of the fuel recently discharged from the core. System modifications are not required to support the PU. The limiting case heat loads at uprate conditions remain bounded by the existing analysis. There is no change to the loss of cooling analysis. The uprate is not expected to have any significant impact on the SFP refueling purification or cooling functions. Therefore, the SFP cooling and purification system is acceptable at the PU conditions. The NRC staff does not expect that the MUR PU will result in a significant change to the operation of the SFPC system. Therefore, the staff concurs with the licensee's conclusion and finds that the SFPC system will not be impacted by the PU.

3.10.2.6 Conclusion

The effect of the MUR PU on containment safety analyses is either bounded by the current containment safety analyses or, in the case of the short-term LOCA, the increase in mass flux is insignificant. Therefore, North Anna Units 1 and 2 remain in compliance with GDCs 16, 38 and 50. The increase of heat loads due to the MUR PU in the containment, control room and on the ESF ventilation systems is insignificant. Therefore, North Anna Units 1 and 2 remain in compliance with GDCs 19 and 60. In summary, the licensee reviewed the design and operation of the plant systems and determined that the proposed MUR PU does not adversely impact any of the systems. For the reasons noted above, the NRC staff concurs with the licensee's conclusion and finds that the plant systems will be acceptable for the MUR PU.

3.11 Changes to Renewed Facility Operating License and TSs

3.11.1 Regulatory Evaluation

This LAR revises the licensed power level identified in Condition 2.C of the Renewed Facility Operating Licenses and the TS definition of reactor thermal power (RTP).

Licensees may revise the TS content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in this section in the context of the specific proposed changes.

3.11.2 Technical Evaluation

The submittal includes TS requirements that would demonstrate compliance with 10 CFR 50.36, "Technical specifications," for plant operating conditions related to the requested authorization for a power level increase. The plant modifications will improve the accuracy of the plant power calorimetric measurement based on the Caldon LEFM Checkplus ™ System (ultrasonic flow meter) instrumentation.

3.11.2.1 Renewed Facility Operating License and Definitions -Rated Thermal Power

The licensee proposed to revise paragraph 2.C of the Renewed Facility Operating License and TS 1.3, Definitions - RTP to reflect the authorized power level increase. The TS RTP will limit the maximum reactor core heat transfer rate to the reactor coolant to 2940 MWt. The NRC staff finds that this change meets 10 CFR 50.36 and is acceptable because the TS limit for operation is derived from the analyses and evaluation included in the safety analysis report (SAR) as accepted by the SE for the requested power level increase discussed herein.

4.0 LICENSEE'S COMMITMENTS

The following table identifies regulatory commitments made by the licensee in the application dated March 26, 2009 and its supplement dated July 8, 2009, and evaluated in this Safety Evaluation. This table has been added to licenses NPF-4 and NPF-7 as License Condition 2.H.

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 1. Dominion will perform the final acceptance of the North Anna Units 1 and 2 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1). | Prior to operating above 2893 MWt (98.4% RTP). |
| 2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0). | Prior to operating above 2893 MWt (98.4% RTP). |
| 3. Procedures and documents for the new UFM (Attachment 5 Section I.1.D.1.1, I.1.H, and VII.2.A). | Prior to operating above 2893 MWt (98.4% RTP). |
| 4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, and VII.2.D). | Prior to operating above 2893 MWt (98.4% RTP). |

| COMMITMENT | SCHEDULED COMPLETION DATE |
|---|---|
| 5. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 6. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4). | Prior to operating above 2893 MWt (98.4% RTP). |
| 7. Replace Steam Generator secondary manway bolts or change cumulative fatigue usage analysis to support using existing bolts for the licensed period for each unit (Attachment 5 Section IV.1.A.vi.2 and IV.1.B.ii). | Prior to exceeding 45 years of in-service use for each secondary manway bolt. |
| 8. The impact of radiation effects on the EQ Program qualification requirements will be determined (Attachment 5 Section V.1.C). | Prior to operating above 2893 MWt (98.4% RTP). |
| 9. The FAC Checkworks SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii). | Prior to operating above 2893 MWt (98.4% RTP). |
| 10. Dominion will determine the EQ-service life of the excore detectors. (Attachment 5 Section II.2). | Prior to operating above 2893 MWt (98.4% RTP). |
| 11. Verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments | Prior to operating above 2893 MWt (98.4% RTP). |
| 12. Confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour completion time | Prior to any such use above 2893 MWt (98.4% RTP). |

## 5.0 STATE CONSULTATION

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

## 6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component 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 (74 FR 23449). 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) 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.

Principal Contributors: J. Parillo, D. Woodyatt, R. Wolfgang, S. Ray, S. Basturescu., R. Lobel, C. Fairbanks, W. Jessup, C. Basavaraju, A. Obodoako, N. Iqbal, V. Sreenivas, K. Martin, T. Mossman, S. Jones, C. Schulten, J. Wallace, L. Ward

Date:

October 22, 2009

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME0965 AND ME0966)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 257 and 238 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated March 26, 2009, as supplemented by letters dated July 8, 16, and 24, 2009.

These amendments revise the licenses and TSs to reflect an increase in the rated thermal power from 2893 megawatts thermal (MWt) to 2940 MWt (1.6 percent increase). The increase is based upon increased feedwater flow measurement accuracy achieved by using high-accuracy Caldon CheckPlus$^{TM}$ Leading Edge Flow Meter ultrasonic flow measurement instrumentation.

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

Sincerely,
*/RA/*
V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 257 to NPF-4
2. Amendment No. 238 to NPF-7
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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Amendment No.: ML092250616 NRR-058 *transmitted by memo dated

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| NAME | JMcHale* | MGavrilas* | MMitchell* | TChan | GWilson* |
| DATE | 07/08/09 | 07/24/09 | 07/30/09 | 10/09/09 | 07/31/09 |
| OFFICE | DE/EICB/BC | DE/EMCB/BC | DIRS/IOLB/BC(A) | DIRS/ITSB/BC | DRA/AADB/BC |
| NAME | WKemper* | MKhanna* | JMunro* | RElliott (css for) | RTaylor* |
| DATE | 07/23/09 | 07/24/09 | 08/05/09 | 10/6/09 | 07/14/09 |
| OFFICE | DRA/AFPB/BC | DSS/SCVB/BC | DSS/SRXB/BC | DSS/SBPB/BC | DSS/SNPB/BC |
| NAME | AKlein* | RDennig* | GCranston* | GCasto | AMendiola |
| DATE | 07/24/09 | 07/27/09 | 07/29/09 | 10/08/09 | 10/13/09 |
| OFFICE | LPL2-1/PM | LPL2-1/LA | OGC NLO/WC | LPL2-1BC | DORL/D |
| NAME | VSreenivas | MO'Brien | LSubin | GKulesa | JGiitter (AHowe for) |
| DATE | 10/20/09 | 10/7/09 w/corrections to SE | 10/20/09 | 10/15/09 | 10/20/09 |

**OFFICIAL RECORD COPY**

Letter to David Heacock from V. Sreenivas

Date: October 22, 2009

AMENDMENT NOS. 257 and 238

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. ME0965 AND ME0966)

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