VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 26, 2009

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 09-033
NLOS/RPC R0
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

Pursuant to 10 CFR 50.90, Dominion requests amendments to Operating Licenses NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. This measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) would increase each unit's rated thermal power (RTP) level from 2893 megawatts thermal (MWt) to 2940 MWt, and make Technical Specifications changes as necessary to support operation at the uprated power level. The proposed change is an increase in RTP of approximately 1.6%.

Dominion developed this LAR utilizing the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." NRC requests for additional information (RAIs) associated with MUR applications for other nuclear units were reviewed for applicability. Information addressing many of those RAIs is included in Attachment 5.

The proposed uprate is characterized as a MUR using the Cameron (formerly Caldon) Leading Edge Flow Meter (LEFM) CheckPlus System to improve plant calorimetric heat balance measurement accuracy.

The proposed changes have been reviewed and approved by the Facility Safety Review Committee.

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Information provided in the attachments to this letter is summarized below:

- Attachment 1 provides Description, Technical Analysis, Regulatory Analysis and Environmental Analysis for the proposed Operating License and Technical Specifications changes. As discussed in this attachment, the proposed amendment does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92. The Facility Safety Review Committee has reviewed and concurred with this determination.

- Attachment 2 contains marked-up pages to reflect the proposed changes to the Operating Licenses and Technical Specifications.

- Attachment 3 contains typed pages to reflect the proposed changes to the Operating Licenses and Technical Specifications.

- Attachment 4 contains marked-up pages to reflect the proposed changes to the Technical Specifications Bases and Technical Requirements Manual. These changes are provided for information only.

- Attachment 5 provides the information recommended to be included in a MUR LAR submittal by NRC RIS 2002-03. This information demonstrates acceptable plant operation at the increased RTP of 2940 MWt.

- Attachment 6 lists the regulatory commitments associated with this LAR.

Additional information required to support the license amendment request that has been determined to be proprietary in accordance with 10 CFR 2.390 is being submitted under separate cover letter (Serial No. 09-033A dated March 27, 2009).

Dominion requests approval of the proposed amendments by January 14, 2010. To permit final installation and operational flexibility, Dominion requests an implementation period from March 14, 2010 to July 14, 2010.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request, with attachments, is being provided to the designated State of Virginia official.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Sincerely,

J. Alan Price
Vice President - Nuclear Engineering

Regulatory commitments are listed in Attachment 6.

Attachments:
1. Discussion of Change.
2. Proposed Operating License and Technical Specifications Pages (Marked-up).
3. Proposed Operating License and Technical Specification Pages (Typed).
4. Proposed Technical Specification Bases and Technical Requirements Manual Pages (Marked-up) For Information Only.
5. NRC Regulatory Issue Summary 2002-03 Requested Information.
6. List of Regulatory Commitments.

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 26TH day of March, 2009.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
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140542
My Commission Expires May 31, 2010

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ATTACHMENT 1

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

DISCUSSION OF CHANGE

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

1.0 DESCRIPTION

Virginia Electric and Power Company (Dominion) proposes a change to the North Anna Power Station (NAPS) Units 1 and 2 Operating Licenses pursuant to 10 CFR 50.90. The measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) would increase each unit's rated thermal power (RTP) from 2893 megawatts thermal (MWt) to 2940 MWt and make Technical Specification changes as necessary to support operation at the uprated power level. The proposed change is an increase in RTP of approximately 1.6%. Unless otherwise noted, 100% power in this LAR refers to 2940 MWt.

This LAR is based on installing and utilizing the Cameron (formally known as Caldon) Leading Edge Flow Meter (LEFM) CheckPlus System as an ultrasonic flow meter (UFM) located in each of the three main feedwater lines supplying the steam generators. The Dominion nomenclature for the Cameron LEFM CheckPlus System is often simplified to feedwater ultrasonic flow meter or UFM.

The original means of measuring feedwater flow using venturis will remain in place performing their original instrumentation protection and control functions. The UFM will be used as the primary method of determining the feedwater flow rate in the plant's calorimetric heat balance and the feedwater venturi based flow rate will become the backup method. The justification for an increase in licensed RTP is based on the increased accuracy of the UFM.

Dominion evaluated the impact of the NAPS MUR uprate to 2940 MWt on applicable systems, structures, components, and safety analyses. Dominion determined that no significant hazards consideration exists as defined by 10 CFR 50.92. In addition, Dominion concluded that the proposed change qualifies for categorical exclusion from performing an environmental assessment as set forth in 10 CFR 51.22(c)(9); therefore, no environmental impact statement or environmental assessment is included or needed for approval.

2.0 BACKGROUND

NAPS was initially licensed to operate at a maximum of 2775 MWt. In Amendments 84 and 71, dated August 25, 1986, the Nuclear Regulatory Commission (NRC) approved NAPS operation at the current power level of 2893 MWt. The proposed MUR power uprate is based on a redistribution of analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed per the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Appendix K originally mandated 102% of licensed power level for light water reactor ECCS evaluation models. The NRC approved a change to the 10 CFR 50, Appendix K requirements on June 1, 2000 effective July 31, 2000. This change provided licensees the option of maintaining the 2% power margin between licensed power level and the ECCS evaluation assumed power level, or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

Implementing the feedwater UFM (Cameron LEFM CheckPlus System) is an effective way to obtain additional plant power without significantly changing current reactor core operations. Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The UFM provides a more accurate measurement of feedwater flow and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.35% at RTP.

The UFM will provide on-line main feedwater flow and temperature measurement to determine reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The UFM consists of a measuring section containing 16 ultrasonic multi-path transit time transducers, one dual resistance temperature detector (RTD), and two pressure transmitters installed in each of the three feedwater lines, and an electronic signal processing cabinet.

The UFM will be used in lieu of the current venturi-based feedwater flow indication and RTD temperature indication to perform the plant calorimetric measurement calculation. The currently installed venturi-based feedwater flow instruments will continue to provide inputs to other indication, protection and control systems, and will be used if the UFM is not functional.

3.0 PROPOSED CHANGE

The proposed (marked-up) Operating License (OL) and Technical Specifications (TS) changes are provided in Attachment 2. The typed OL and TS pages are provided in Attachment 3.

The proposed OL and TS changes are described below.

<u>Operating License - Maximum Power Level</u>

Paragraph 2.C(1), "Maximum Power Level," of the Unit 1 and Unit 2 Operating Licenses (NPF-4 and NPF-7 respectively) authorizes facility operation at a reactor core power level not in excess of 2893 megawatts

(thermal). The proposed change increases the Maximum Power Level from its current value of 2893 MWt to 2940 MWt.

TS Section 1.1, Definitions – Rated Thermal Power

The Technical Specification definition of Rated Thermal Power (RTP) limits the reactor core power level to 2893 MWt. The MUR power uprate is equivalent to an approximately 1.6% increase in the current RTP. The RTP definition is revised to change the value from 2893 MWt to 2940 MWt, to be consistent with the Maximum Power Level in Facility Operating License Paragraph 2.C(1).

TS Bases and Technical Requirements Manual

TS Bases and Technical Requirements Manual (TRM) changes are being made to support this LAR. These changes include: TS Bases Changes to RTS Instrumentation, Section B 3.3.1; Main Steam Safety Valve (MSSV), Section B 3.7.1; Emergency Condensate Storage Tank (ECST), Section B 3.7.6; and the addition of a new TRM section for UFM. The TS Bases and TRM changes are provided for information only in Attachment 4. The TRM is incorporated by reference in Updated Final Safety Analysis Report (UFSAR) Section 16.2. As stated in UFSAR Section 16.2, TRM changes are controlled using the 10 CFR 50.59 process.

Updated Final Safety Analysis Report

Changes to the NAPS UFSAR are being made to support this LAR. These changes will be made in accordance with 10 CFR 50.59.

4.0 TECHNICAL ANALYSIS

NAPS Units 1 and 2 are presently licensed for a RTP of 2893 MWt. Using more accurate feedwater flow measurement equipment supports an approximately 1.6% increase to 2940 MWt. The power uprate evaluations addressed the following categories: nuclear steam supply system (NSSS) performance parameters, accidents, design transients, systems, components, nuclear fuel, and interfaces between NSSS and balance-of-plant (BOP) systems. The evaluation conclusions are summarized in Attachment 5, information requested in NRC Regulatory Issue Summary (RIS) 2002-03. These analyses were reviewed to provide assurances that they remain bounding for the proposed power uprate. Non-bounding analyses are discussed in Attachment 5, Section III.

Table 4.0-1 indicates the power levels used for the NAPS MUR power uprate analyses and evaluations. Each area of analysis scope assumed an appropriate core power that bounds the proposed 2940 MWt value (nominal or nominal plus uncertainty).

Table 4.0-1
Analysis Power Levels for North Anna Units 1 and 2 MUR Uprating

Analysis Scope	Core Power MWt	NSSS Power MWt[4]	Source
NSSS	2956[1]	2968	NSSS Design Parameters
Safety Analyses	2951[2]	2963	UFSAR Chapters 6 and 15
Statistical DNBR Events	2942.2[3]	2955	UFSAR Chapter 15
Safety-Related System Evaluations	2951[2]	2963	Consistent with UFSAR safety analyses
BOP System Evaluations	2942.2[3]	2955	

1. 102% of current analyzed core power of 2898 MWt
2. 102% of current RTP of 2893 MWt
3. 101.7% of current RTP of 2893 MWt
4. The analyses use 12 MWt for RCP net heat addition. For the BOP system evaluations, the NSSS power is rounded up to the next whole number

The NSSS design thermal and hydraulic parameters derived from the power uprate conditions serve as the basis for the NSSS analyses. A detailed review of the accident analyses was performed for the steam generator tube rupture, loss of coolant accident (LOCA), and non-LOCA areas. The currently assumed loss of coolant mass and energy release remains bounding. The radiological consequence evaluation is bounded by the current analysis since the radiological source term has not increased. The fuel was evaluated for its ability to perform at the uprated power level. Dominion concludes that the changes to the NAPS design basis and transient analyses are acceptable. Each of the NSSS systems and components were evaluated at the uprated conditions. The BOP systems, electrical power systems, control systems and instrumentation systems were also evaluated at the uprated conditions. The analyses and evaluations performed demonstrate that the acceptance criteria continue to be met. NAPS Units 1 and 2 require minimal plant modifications to safely operate at the uprated conditions (Attachment 5 Section VII.2.B).

4.1 Nuclear Steam Supply System Design Parameters

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. The design parameters are established using conservative input assumptions to provide bounding conditions used in the NSSS analyses. They provide the primary and secondary side system conditions (thermal power, temperatures, pressures, flow) that are used as the basis for the NSSS analyses and evaluations. These parameters were revised to account for the increase in analyzed core power from 2898 MWt to 2956 MWt. The new parameters are listed in Table 4.0-2. These parameters have been incorporated, as required, into the applicable NSSS system and component evaluations and safety analyses performed to support the power uprate.

4.2 Input Parameters

The major input parameters used to calculate the four cases of NSSS design parameters are as follows:

- NSSS uprated power level of 2968 MWt (2956 MWt core power plus 12 MWt reactor coolant pump (RCP) net heat input).
- Core bypass flow of 6.5%, which accounts for thimble plug removal.
- Feedwater temperature of 449.0°F.
- Westinghouse Model 54F replacement steam generators (SG).
- Vessel average temperature (T_{avg}) range of 580.8°F to 586.8°F.
- Maximum steam generator moisture carryover of 0.10%.
- Steam generator tube plugging (SGTP) levels of 0% and 7%.
- Thermal design flow maintained at 92,800 gpm/loop.
- Reactor coolant pressure of 2250 psia, which is the current operating value.

4.3 Parameter Cases

Four cases of NSSS design parameters were used to evaluate the power uprate impact. These four cases are shown in Table 4.0-2.

Cases 1 and 2 represent parameters applicable to most NSSS analyses that are based on the minimum T_{avg} of 580.8°F. Case 2 is based on an average 7% SGTP and yields the minimum SG secondary side steam pressure and temperature. Note that the primary side temperatures are identical for these two cases.

Cases 3 and 4 represent parameters applicable to most NSSS analyses that are based on the maximum T_{avg} of 586.8°F. Case 3 is based on an average 0% SGTP and yields the maximum SG secondary side steam pressure and temperature. Note that the primary side temperatures are identical for these two cases.

The various NSSS analyses and evaluations (e.g., systems, components and materials) performed for the MUR power uprate incorporated the design parameters appropriate for those analytical areas.

Table 4.0-2
NSSS Design Parameters for North Anna Units 1 and 2 MUR Uprating

Parameter	Current Design Conditions	Bounding MUR 2% Uprate			
		Case 1	Case 2	Case 3	Case 4
THERMAL DESIGN					
NSSS Power, %	100	102	102	102	102
MWt	2910(2)	2968	2968	2968	2968
10^6 BTU/hr	9929	10,127	10,127	10,127	10,127
Reactor Power, MWt	2898	2956	2956	2956	2956
10^6 BTU/hr	9888	10,086	10,086	10,086	10,086
Thermal Design Flow, Loop gpm	92,800	92,800	92,800	92,800	92,800
Reactor 10^6 lb/hr	104.3	105.3	105.3	104.4	104.4
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.5(1)	6.5(1)	6.5(1)	6.5(1)	6.5(1)

Table 4.0-2 (Continued)
NSSS Design Parameters for North Anna Units 1 and 2 MUR Uprating

Parameter	Current Design Conditions	Bounding MUR 2% Uprate			
		Case 1	Case 2	Case 3	Case 4
Reactor Coolant Temperature, °F					
Core Outlet	624.0	620.5	620.5	626.1	626.1
Vessel Outlet	621.2	616.3	616.3	621.9	621.9
Core Average	590.4	585.2	585.2	591.3	591.3
Vessel Average	586.8	580.8	580.8	586.8	586.8
Vessel/Core Inlet	552.3	545.4	545.4	551.7	551.7
Steam Generator Outlet	552.0	545.0	545.0	551.4	551.4
Steam Generator					
Steam Temperature, °F	525.2	521.2	518.8	527.8	525.4
Steam Pressure, psia	850	821	804	869	851
Steam Flow, 10^6 lb/hr total	12.78	13.17	13.16	13.20	13.19
Feedwater Temperature, °F	440.0	449.0	449.0	449.0	449.0
Moisture, % maximum	0.10	0.10	0.10	0.10	0.10
Steam Generator Tube Plugging, %	7	0	7	0	7
Zero Load Temperature, °F	547	547	547	547	547
HYDRAULIC DESIGN					
Mechanical Design Flow, gpm	105,200	105,200	105,200	105,200	105,200

1. Core bypass flow covers thimble plug removal for Units 1 & 2, upflow conversion for Unit 1, and current downflow configuration for Unit 2
2. This represents the current NSSS analyzed power level of 2898 MWt core power plus 12 MWt for RCP net heat input

5.0 REGULATORY ANALYSIS

Dominion has evaluated the License Amendment Request (LAR) against the 10 CFR 50.92 criteria to determine if any significant hazards consideration is involved. Dominion has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change will increase the North Anna Power Station (NAPS) Units 1 and 2 rated thermal power (RTP) from 2893 megawatts thermal (MWt) to 2940 MWt. Nuclear steam supply system and balance-of-plant systems, components and analyses that could be affected by the proposed change to the RTP were evaluated using revised design parameters. The evaluations determined that these structures, systems and components are capable of performing their design function at the proposed uprated RTP of 2940 MWt. An evaluation of the accident analyses demonstrates that the applicable analysis acceptance criteria are still met with the proposed changes. Power level is an input assumption to equipment design and accident analyses, but it is not a transient or accident initiator. Accident initiators are not affected by the power uprate, and plant safety barrier challenges are not created by the proposed changes.

The radiological consequences of operation at the uprated power conditions have been assessed. The proposed change to RTP does not affect release paths, frequency of release, or the analyzed source term for any accidents previously evaluated in the NAPS Updated Final Safety Analysis Report. Structures, systems and components required to mitigate transients are capable of performing their design functions with the proposed changes, and are thus acceptable. Analyses performed to assess the effects of mass and energy releases remain valid. The source term used to assess radiological consequences was reviewed and determined to bound operation at the proposed power level.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any proposed changes. The UFM has been analyzed, and system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems and components previously required for transient mitigation are still capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at RTP of 2940 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Core analyses demonstrate that power uprate implementation will continue to meet the current nuclear design basis. Impacts to components associated with the reactor coolant system pressure boundary structural integrity, and factors such as pressure-temperature limits, vessel fluence, and pressurized thermal shock were determined to be bounded by the current analyses.

   Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. The current NAPS safety analyses, including the design basis radiological accident dose calculations, bound the power uprate.

   Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above, Dominion concludes that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and a finding of no significant hazards consideration is acceptable.

6.0 ENVIRONMENTAL ANALYSIS

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

Dominion has concluded that this license amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment is required in connection with issuance of the proposed license amendment. This determination is based on the following:

(i) The license amendment request does not involve a significant hazards consideration, as described in the significant hazards evaluation.

(ii) The proposed change does not involve installing new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite.

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level.

Gaseous and liquid radwaste effluent activity is expected to increase from current levels by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and10 CFR 50, Appendix I limits per the NAPS Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

Solid radwaste effluent activity is expected to increase from current levels proportionately to the increase in long half-life coolant activity. The total long-lived activity is bounded by the percent of power uprate. Changes in solid waste volume are not expected.

Therefore, the proposed license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

(iii) The license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage of core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within acceptable limits by the site Radiation Protection Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

7.0 PRECEDENT

License amendment applications based on the Cameron (formerly Caldon) LEFM CheckPlus system were previously approved for PWRs Seabrook Station (Reference 8.1), Crystal River 3 (Reference 8.2) and Vogtle 1 & 2 (Reference 8.3). These submittals requested NRC approval to increase licensed power level by reducing uncertainty through the use of the LEFM CheckPlus system for feedwater flow measurement. The North Anna Units 1 & 2 submittal is comparable to those license amendment requests.

8.0 REFERENCES

8.1 NRC letter to FPL Energy Seabrook, *Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML061360034, May 22, 2006.

8.2 NRC letter to Crystal River 3, *Crystal River 3 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML073600419, December 26, 2007.

8.3 NRC letter to Vogtle Electric Generating Plant, Units 1 and 2, *Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate*, ML080350347, February 27, 2008.

ATTACHMENT 2

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES (MARKED-UP)

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

(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 2893 megawatts (thermal).

2940

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2893 megawatts (thermal).

(2) <u>Technical Specifications</u> 2940

The Technical Specifications contained in Appendix A, as revised through Amendment No. 234 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Additional Conditions</u>

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

1.1 Definitions

PHYSICS TESTS (continued)

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

c. Otherwise approved by the Nuclear Regulatory Commission.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt. 2940

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:

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

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

ATTACHMENT 3

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES (TYPED)

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

(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. , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(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 No. 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

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 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: 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 b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

ATTACHMENT 4

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED TECHNICAL SPECIFICATIONS BASES AND TECHNICAL REQUIREMENTS MANUAL PAGES (MARKED-UP)

FOR INFORMATION ONLY

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output every 24 hours. If the calorimetric heat balance calculation results exceeds the power range channel output by more than +2% RTP, the power range channel is not declared inoperable, but must be adjusted. The power range channel output shall be adjusted consistent with the calorimetric heat balance calculation results if the calorimetric calculation exceeds the power range channel output by more than +2% RTP. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

If the calorimetric is performed at part power (< 85% RTP), adjusting the power range channel indication in the increasing power direction will assure a reactor trip below the safety analysis limit (< 118% RTP). Making no adjustment to the power range channel in the decreasing power direction due to a part power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation power is less than the power range channel output. To provide close agreement between indicated power and to preserve operating margin, the power range channels are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range channel output is adjusted in the decreasing power direction due to a part power calorimetric (< 85% RTP). This action may introduce a non-conservative bias at higher power levels which may result in an NIS reactor trip above the safety analysis limit (> 118% RTP). The cause of the non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is typically a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the ΔP error has not changed. An evaluation of extended operation at part power conditions would conclude that it is prudent to administratively adjust the setpoint of the Power

(continued)

The ultrasonic flow meter provides more accurate feedwater flow measurement than the existing venturis. Feedwater flow measurement from the ultrasonic flow meter may be used to compute the secondary side power calorimetric. If feedwater ultrasonic flow meter data is used for the calorimetric at reduced flow, the accuracy is also reduced however not as significantly as with the feedwater venturi data.

BASES

APPLICABLE SAFETY ANALYSES (continued)

if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

100.37 %

LCO

The accident analysis requires five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

<u>A.1</u>

In the case of only a single inoperable MSSV on one or more steam generators, when the MTC is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference ~~6~~ 5, with an appropriate allowance for calorimetric power uncertainty.

<u>B.1 and B.2</u>

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators when the MTC is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

remaining OPERABLE MSSVs is insufficient. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference ~~6~~ 5, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Protection System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

- INFO ONLY -

NO CHANGES ON THIS PAGE

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

SRs are specified in the Inservice Testing Program. MSSVs are to be tested in accordance with the requirements of the ASME Code (Ref. 4) which provides the activities and frequencies necessary to satisfy the SR. The MSSV lift settings given in the LCO are for operability, however, the valves are reset to ±1% during the surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure.

REFERENCES

1. UFSAR, Section 10.3.1.
2. ASME, Boiler and Pressure Vessel Code, Section III.
3. UFSAR, Section 15.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures accommodated by the accident include the following:

a. Failure of the diesel generator powering the motor driven AFW pump to one unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in ECST inventory determinations is a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Engineered Safety Features Actuation System (LCO 3.3.2, ESFAS) starts the AFW system and would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The ECST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

100.37%

LCO

To satisfy accident analysis assumptions, the ECST must contain sufficient cooling water to remove decay heat for 30 minutes following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The ECST level required is equivalent to a contained volume of ≥ 110,000 gallons, which is based on holding the unit in MODE 3 for 8 hours, or maintaining the unit in MODE 3 for 2 hours followed by a 4 hour cooldown to RHR entry

(continued)

3.3 INSTRUMENTATION

3.3.10 Feedwater Ultrasonic Flow Meter Calorimetric

TR 3.3.10 The Feedwater Ultrasonic Flow Meter (UFM) Calorimetric shall be FUNCTIONAL with:

a. The Feedwater UFM System FUNCTIONAL.

b. The Plant Computer System (PCS) calorimetric program FUNCTIONAL.

APPLICABILITY: MODE 1 with THERMAL POWER > 2893 MWt (98.4% RTP).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Feedwater UFM System not FUNCTIONAL.	A.1 Change the calorimetric program from the Feedwater UFM System to the Normalized Feedwater Venturi System. AND A.2 Restore Feedwater UFM System to FUNCTIONAL status.	1 hour 48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤2893 MWt (98.4% RTP). AND B.2 Change the calorimetric program from the Normalized Feedwater Venturi System to the Feedwater Venturi System.	1 hour 1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. PCS calorimetric program not FUNCTIONAL for reasons other than Condition A.	C.1 Verify THERMAL POWER ≤2940 MWt (100% RTP) by monitoring alternate power indications.	Immediately
	AND	
	C.2.1 Restore the PCS calorimetric program to FUNCTIONAL status.	Prior to performing the next required power range channel calorimetric heat balance comparison per TS SR 3.3.1.2
	OR	
	C.2.2 Reduce THERMAL POWER to ≤2893 MWt (98.4% RTP) by monitoring alternate power indications.	Prior to performing the next required power range channel calorimetric heat balance comparison per TS SR 3.3.1.2

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.10.1	-------------------NOTE-------------------- When in Condition A, the Normalized Feedwater Venturi System will be used to perform the 24 hour surveillance. --- Perform TS SR 3.3.1.2 using the Feedwater UFM System.	Prior to exceeding 2893 MWt (98.4% RTP) <u>AND</u> Once per 24 hours thereafter
TSR 3.3.10.2	Perform Channel Calibration of the Feedwater UFM System instrumentation.	Once per 18 months

B 3.3 INSTRUMENTATION

B 3.3.10 Feedwater Ultrasonic Flow Meter Calorimetric

BASES

BACKGROUND AND APPLICABLE SAFETY ANALYSES

North Anna was initially licensed to operate at a maximum reactor power level of 2775 (MWt). In August 1986, the Nuclear Regulatory Commission (NRC) approved NAPS operation at the reactor power level of 2893 MWt. A second power uprate to a reactor power level of 2940 MWt is based on a redistribution of analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed per the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Appendix K originally mandated 102% of licensed power level for light water reactor ECCS evaluation models. The NRC approved a change to the 10 CFR 50, Appendix K requirements on June 1, 2000. This change provided licensees the option of maintaining the 2% power margin between the licensed power level and the ECCS evaluation assumed power level, or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The Feedwater Ultrasonic Flow Meter (UFM) System provides a more accurate measurement of feedwater flow compared to the feedwater venturis and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty that is less than the accident analysis allowance of 0.37% RTP. The remaining approximately 1.6% RTP margin is the basis for the power uprate. This type of power uprate is referred to as a Measurement Uncertainty Recapture (MUR) Uprate.

The Feedwater UFM System provides on-line main feedwater flow and temperature measurements to determine reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The system measures the transit times of ultrasonic pulses traveling through the flowing fluid. Sound travels faster when the pulse traverses the pipe with the flow and slower against the flow due to the doppler effect. The system uses these transit times and time differences between pulses to

(continued)

BASES

BACKGROUND AND APPLICABLE SAFETY ANALYSES (continued)	determine the fluid velocity. The system also measures the speed of sound in water and uses this measurement to determine the feedwater temperature. The Feedwater UFM System consists of an electronic processing cabinet installed in the Cable Spreading Room, and measurement spool pieces installed in each of the three main Feedwater flow lines. Each measurement spool piece consists of 16 ultrasonic, multi-path, transit time transducers, one dual resistance temperature detector (RTD), and two pressure transmitters. The 16 transducers are separated into two planes, four paths in each plane. Each plane provides input to its own subsystem of electronic hardware. Each transducer can be removed without disturbing the pressure boundary. The electronics for the two subsystems, while electrically separated, are housed in a single processing cabinet installed in the Cable Spreading Room. The Feedwater UFM System performs on-line self diagnostics to verify system operation within design basis uncertainty limits. Any out of specification condition will result in a control room annunciator. A failure between the Feedwater UFM System electronics cabinet and the plant computer will also result in a control room annunciator. If the feedwater UFM failure annunciator is received, the Feedwater UFM System will be declared not functional. The control room annunciator response procedure provides guidance to the operators for initial alarm diagnosis and response. Although a single plane malfunction results in a minimal increase in feedwater flow uncertainty, operators will conservatively respond to a single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode. The Feedwater UFM System will be used in place of the venturi-based feedwater flow and RTD temperature instrumentation to perform the calorimetric calculation. The venturi-based feedwater flow instruments will continue to provide inputs to other indications, protection and control systems, and will be used if the Feedwater UFM System is not functional.
LCO	The Feedwater UFM calorimetric requires the Feedwater UFM System and the Plant Computer System (PCS) calorimetric program to be FUNCTIONAL.

BASES

APPLICABILITY

In MODE 1 with THERMAL POWER > 2893 MWt (98.4% RTP), the Feedwater UFM calorimetric must be FUNCTIONAL. The Feedwater UFM calorimetric provides a more accurate measurement of reactor thermal power than the feedwater venturi-based calorimetric. The improved accuracy of the Feedwater UFM calorimetric is the basis for operating above 2893 MWt (98.4% RTP).

ACTIONS

A.1 and A.2

With the Feedwater UFM System nonfunctional, action must be taken to restore FUNCTIONAL status in 48 hours, provided THERMAL POWER remains above 2893 MWt (98.4% RTP). If the Feedwater UFM System is not returned to service in 48 hours, reactor power is required to be reduced to ≤2893 MWt (98.4% RTP).

The Normalized Feedwater Venturi System calorimetric is used during the 48 hour completion time when the Feedwater UFM System is nonfunctional. The Normalized Feedwater Venturi System calorimetric receives input from the feedwater venturis for feedwater flow rate calculation. The feedwater flow from the three venturis is normalized to the Feedwater UFM System flow rate. In addition, the feedwater temperature and feedwater pressure data is normalized to the more accurate data from the Feedwater UFM System. Normalization of data results in the Normalized Feedwater Venturi System calorimetric closely matching the Feedwater UFM System calorimetric.

The accuracy of the instruments used to perform the Normalized Feedwater Venturi System calorimetric will not significantly change over 48 hours. As a result, significant calorimetric measurement uncertainty will not occur over a 48 hour period. The 1 hour completion time to change the calorimetric program from the Feedwater UFM System to the Normalized Feedwater Venturi System is reasonable based on operating experience.

During the 48-hour COMPLETION TIME, if THERMAL POWER is reduced to ≤2893 MWt (98.4% RTP), THERMAL POWER cannot be increased to > 2893 MWt (98.4% RTP) until a calorimetric is performed using the Feedwater UFM System in accordance with TSR 3.3.10.1. This restriction is required to ensure that the plant transient has not affected the accuracy of the Normalized Feedwater Venturi System.

BASES

ACTIONS (continued)

B.1 and B.2

If the Feedwater UFM System cannot be restored to functional status within the associated completion time, the unit must be placed in a condition in which the LCO does not apply. To achieve this status, reactor power must be reduced to ≤2893 MWt (98.4% RTP). The 1 hour completion time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The accuracy of the Normalized Feedwater Venturi calorimetric program can be impacted over time. For this reason, the calorimetric program will be changed from the Normalized Feedwater Venturi System to the Feedwater Venturi System after a 48-hour time period. The 1-hour completion time to change the calorimetric program from the Normalized Feedwater Venturi System to the Feedwater Venturi System is reasonable based on operating experience.

C.1, C.2.1 and C.2.2

A failure of the PCS calorimetric program would result in the loss of computer generated calorimetric programs. In this case, THERMAL POWER would be determined by monitoring alternate power indications using the power range nuclear instrumentation (NIs) and RCS loop ΔTs. The procedure for loss of the PCS provides guidance for monitoring reactor power.

Operation at 100% RTP may continue until the next required performance of TS SR 3.3.1.2, "Calorimetric Heat Balance Calculation." If the computer calorimetric program is nonfunctional, a manual calorimetric heat balance calculation would be required to meet the requirements of TS SR 3.3.1.2. The manual calorimetric heat balance calculation uses data from the feedwater venturis, not the Feedwater UFM System. Therefore, the manual calorimetric cannot be used to satisfy the surveillance requirement when operating above 2893 MWt (98.4% RTP).

If the PCS calorimetric program is not restored to FUNCTIONAL status prior to the performance of the next calorimetric required by TS SR 3.3.1.2, THERMAL POWER would be reduced to ≤2893 MWt (98.4% RTP) and a manual calorimetric would be performed. The power reduction and performance of a manual calorimetric would have to be

(continued)

BASES

ACTIONS

C.1, C.2.1 and C.2.2 (continued)

completed within the surveillance interval required by TS SR 3.3.1.2. Thermal power would be reduced by monitoring alternate power indications using the power range nuclear instrumentation (NIs) and RCS Loop ΔTs.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Note 1 has been added to clarify that when in Condition A, the Normalized Feedwater Venturi System will be used to perform the required 24 hour surveillance.

This SR ensures that a calorimetric using the more accurate measurements of feedwater flow from the Feedwater UFM System is performed prior to exceeding a THERMAL POWER level of 2893 MWt (98.4% RTP). The Feedwater UFM System is used to perform the TS SR 3.3.1.2 surveillance once per 24 hours thereafter.

If THERMAL POWER is reduced to ≤2893 MWt (98.4% RTP), a calorimetric using the Feedwater UFM System must be performed prior to exceeding 2893 MWt (98.4% RTP). This initial surveillance is required to be performed even if power is reduced for a short period of time and a calorimetric using the Feedwater UFM System had been performed within the previous 24 hours.

A calorimetric using the Feedwater UFM System is required to be performed each time power will be increased > 2893 MWt (98.4% RTP). This ensures the requirements (feedwater UFM calorimetric) are met for operating at a power level of > 2893 MWt (98.4% RTP).

A channel calibration of the Feedwater UFM System instrumentation is performed every 18 months. The Feedwater UFM System instrumentation calibration procedure and frequency of calibration, are based on vendor recommendations.

REFERENCES

1. License Amendment Request – Measurement Uncertainty Recapture Power Uprate.

2. TS SR 3.3.1.2.

ATTACHMENT 5

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

NRC REGULATORY ISSUE SUMMARY 2002-03
REQUESTED INFORMATION

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

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Acronym List

Expression	Definition or Use
AC	alternating current
AFW	auxiliary feedwater system
ALARA	as low as reasonably achievable
AMBW	Advanced Mark-BW
AMSAC	ATWS mitigation system actuation circuitry
AOV	air operated valve
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BOP	balance of plant
B&PV	boiler and pressure vessel
CCW	component cooling water
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPU	central processing unit
CRDM	control rod drive mechanism
DC	direct current
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
ECST	emergency condensate storage tank
EDG	emergency diesel generator
EFPY	effective full power years
EOL	end of life
EOP	emergency operating procedure
EQ	environmental qualification (10 CFR 50.49)
ESF	engineered safety features
FAC	flow accelerated corrosion
FW	feedwater
GL	NRC Generic Letter
HELB	high energy line break
HP	horse power
ISI	inservice inspection

Acronym List

Expression	Definition or Use
IST	inservice testing
LAR	license amendment request
LEFM	leading edge flow meter
LOCA	loss of coolant accident
LOOP	loss of offsite power
MOV	motor operated valve
MSLB	main steam line break
MSS	main steam system
MSSV	main steam safety valve
MSTV	main steam trip valve
MUR	measurement uncertainty recapture
MWD/MTU	megawatt day per metric ton uranium
MWe	megawatt electric
MWt	megawatt thermal
NAIF	North Anna Improved Fuel
NAPS	North Anna Power Station
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PCT	peak clad temperature
PORV	power operated relief valve
PTS	pressurized thermal shock (10 CFR 50.61)
PWR	pressurized water reactor
QS	quench spray
RCCA	rod control cluster assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal system
RS	recirculation spray
RT_{NDT}	reference temperature nil ductility transition
RT_{PTS}	reference temperature pressurized thermal shock
RTD	resistance temperature detector
RTP	rated thermal power
RTS	reactor trip system
RWST	refueling water storage tank

Acronym List

Expression	Definition or Use
SBO	station blackout
SBLOCA	small break loss of coolant accident
SER	safety evaluation report
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SW	service water
TRM	Technical Requirements Manual
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UFM	ultrasonic flow meter
VCT	volume control tank
VEPCO	Virginia Electric and Power Company
X/Q	radiological atmospheric dispersion factor

Introduction

This attachment contains the Dominion responses to the NRC Regulatory Issue Summary 2002-03, requested information for MUR power uprates. The LAR attachment sections match the NRC Regulatory Issue Summary 2002-03, sections for ease of review.

I FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

1. A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. The description should include:

   A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

   B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

   C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

   D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

   E. A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty

   F. Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

      i. maintaining calibration

      ii. controlling software and hardware configuration

      iii. performing corrective actions

      iv. reporting deficiencies to the manufacturer

v. receiving and addressing manufacturer deficiency reports

G. A proposed allowed outage time for the instrument, along with the technical basis for the time selected

H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

RESPONSE TO I – FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

I.1 Detailed Description of the North Anna Units 1 and 2 Implementation of the Feedwater Ultrasonic Flow Meter

The NAPS feedwater ultrasonic flow meter is a Cameron LEFM CheckPlus ultrasonic multi-path, transit time flowmeter. This equipment also provides a highly accurate feedwater temperature that will be input to the heat balance. This advanced flow measurement system design is described in detail by the manufacturer, Cameron Inc. (formerly Caldon), in Topical Reports ER-80P, Revision 0 (Reference I-1), and ER-157P, Revision 5 (Reference I-2).

The LEFM CheckPlus system consists of an electronic cabinet installed in the Cable Spreading Room, and measurement spool pieces installed in each of the three main feedwater flow lines between the existing feedwater venturi flow meters and the main feedwater check valves. The spool pieces are installed well downstream of the existing feedwater flow venturis, and will have no impact on venturi performance. The UFMs were calibrated at the Alden Research Laboratory facility using the current plant piping configuration and variations of the plant configuration. The calibration test determines the meter calibration constant, or meter factor. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests are performed to determine meter factor sensitivity to upstream hydraulics.

Each measurement section consists of 16 ultrasonic, multi-path, transit time transducers, one dual resistance temperature detector (RTD), and two pressure transmitters. Each transducer may be removed at full power conditions without disturbing the pressure boundary. These flow elements conform to the installation location requirements specified in Topical Reports ER-80P and ER-157P.

The UFM measures the transit times of ultrasonic energy pulses traveling along chordal acoustic paths through the flowing fluid. This technology provides higher accuracy and reliability than the existing flow instruments. Sound travels faster when the pulse traverses the pipe with the flow and slower against the flow due to the Doppler effect. The UFM uses these transit times and time differences between pulses to determine the fluid velocity. The UFM also measures the speed of sound in water and uses this measurement to determine the feedwater temperature.

The electronic cabinet controls magnitude and sequences transducer operation; makes time measurements; and calculates volume, temperature and mass flow. The system software employs the ultrasonic transit time method to measure velocities at precise locations. The system numerically integrates the measured velocities. The system software has been developed and maintained under a verification and validation program. The verification and validation program has been applied to the system software and hardware, and includes a detailed code review. The feedwater mass flow rate and temperature are displayed on the electronic cabinet and transmitted to the plant process computer for use in the calorimetric measurement (secondary plant energy balance) of reactor thermal output. The system will utilize continuous calorimetric power determination by direct, redundant links with the plant computer, and will incorporate self-verification features. These features ensure that system performance is consistent with the design basis.

The system has two modes of operation: Normal operation and Maintenance mode. Normal operation is defined as CheckPlus operation. In this mode, both planes of transducers are in service and system operations are processed by both CPUs. If the system is subjected to a failure involving a transducer, failure of one plane of operation or if a central processing unit (CPU) related malfunction occurs, the system reverts to the Check system or Maintenance mode. When a plane of operation is lost, the system alerts the control room operators through the annunciator window for Feedwater Ultrasonic Flow Meter Failure, and shifts from Normal operation to Maintenance mode. If the system suffers a loss of AC power or other total failure, the system also alerts the operators through the aforementioned annunciator. Operations personnel are also alerted to system trouble through annunciator window for Feedwater Ultrasonic Flowmeter Trouble if the electronic cabinet internal temperature is high or when other trouble conditions occur as determined by the plant computer.

The improved measurement accuracy for feedwater mass flow and temperature and a change in the way instrument uncertainty is combined for other parameters (e.g., steam temperature) results in a total uncertainty of 0.35% at RTP. This is more accurate than the nominal 2% RTP used in the accident analyses or the uncertainty currently obtainable with precision, venturi-based instrumentation and RTDs.

The UFM indications of feedwater mass flow and temperature will be directly substituted for the existing venturi-based flow and RTD temperature inputs currently used in the plant calorimetric measurement calculations. The plant computer system calorimetric programs will be revised to receive data from the UFM and from loop-specific, high-capacity SG blowdown flow, to calculate UFM mode calorimetric power, and to calculate feedwater and steam venturi flow calorimetric power using feedwater flow, temperature and pressure values normalized to UFM values. The existing venturi-based feedwater flow and RTD temperature will continue to be used for feedwater control and other functions, and may be used for plant calorimetric measurement calculations when the UFM is unavailable.

I.1.A Cameron Topical Reports Applicable to the LEFM CheckPlus System

The referenced Topical Reports are:

ER-80P, Rev. 0 (Reference I-1)

ER-157P, Rev. 5 (Reference I-2)

I.1.B NRC Approval of Cameron LEFM CheckPlus System Topical Reports

The NRC approved the Topical Reports referenced in I.1.A above on the following dates:

ER-80P, NRC SER dated March 8, 1999 (Reference I-3)

ER-157P, NRC SER dated December 20, 2001 (Reference I-4)

The NRC performed additional evaluations on the acceptability of the Cameron LEFM. The evaluation results are documented in Reference I-5, which addressed the hydraulic aspects of Cameron LEFMs in response to industry operating experience. The NRC staff concluded that the Cameron LEFM Check and CheckPlus performance was consistent with the Cameron Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff (Reference I-5).

I.1.C North Anna Power Station (NAPS) Implementation of Guidelines and NRC SER for the Cameron LEFM CheckPlus System

The LEFM CheckPlus system is permanently installed per the requirements specified in Topical Reports ER-80P and ER-157P. The system will be used for continuous calorimetric power determination by direct, redundant links with the plant computer. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The plant computer system software continuously adjusts the venturi flow coefficients and the feedwater RTD temperatures to the more accurate UFM values. The feedwater flow values for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow are normalized to equal the UFM feedwater flow. Normalization is performed on a loop basis. The feedwater temperature values used to determine densities for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow are based on normalized feedwater RTD temperatures biased to equal the UFM feedwater temperatures. Feedwater pressure measurements will be normalized to the more accurate data from the UFM. Additionally, the feedwater venturis were recently recalibrated at Alden Research Labs.

The NAPS LEFM CheckPlus system was calibrated in a site-specific model test at Alden Research Labs, with traceability to National Standards. A copy of the Alden Research Labs certified calibration report is contained in the Cameron Meter Factor Reports. The LEFM CheckPlus system installation and commissioning is performed according to Cameron procedures. These procedures include verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those during site-specific model testing.

I.1.D Disposition of NRC SER Criteria During Installation

In approving Cameron Topical Reports ER-80P and ER-157P, the NRC established four criteria each licensee must address. The four criteria are listed below along with a discussion of how each will be satisfied for NAPS Units 1 and 2.

I.1.D.1 NRC Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the UFM. These procedures should include processes and contingencies for a not functional UFM and the effect on thermal power measurement and plant operation.

I.1.D.1.1 Response to NRC Criterion 1

Power uprate license amendment implementation will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing and training at the uprated power level using the LEFM CheckPlus system. A preventive maintenance program will be developed for the UFM using the vendor's maintenance and troubleshooting manual. Work on the UFM will be performed by site instrumentation and control personnel qualified per the NAPS Instrumentation & Control Training Program.

The preventive maintenance activities include:

- General terminal and cleanliness inspection
- Power supply inspection
- Central Processing Unit inspection

- Acoustic Processor Unit checks
- Analog input/output checks
- Alarm Relay checks
- Watchdog Timer checks that ensure the software is running
- Communication checks
- Transducer checks
- Calibration checks on each feedwater pressure transmitter

The preventive maintenance program and UFM continuous self-monitoring ensure that the UFM remains bounded by the Topical Report ER-80P analysis and assumptions. Establishing and continued adherence to these requirements assures that the UFM system is properly maintained and calibrated.

Contingency plans for plant operation with the UFM not functional are described in Sections I.1.G and I.1.H below.

I.1.D.2 NRC Criterion 2

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

I.1.D.2.1 Response to NRC Criterion 2

Because the UFM installation will not be completed at NAPS Unit 1 until the Spring 2009, the Criterion 2 does not apply to NAPS Unit 1.

UFMs were installed in NAPS Unit 2 during the Fall 2008 Refueling Outage, with commissioning and calibration completed on December 8, 2008. Active monitoring has been ongoing since that time. The UFM feedwater flow and temperature data have been compared to the feedwater flow venturis output and the feedwater RTD output. The data comparison demonstrated that the UFM is consistent with the venturi feedwater flow and RTD feedwater temperature. There have been no maintenance related activities since UFM installation. The NAPS Unit 2 UFMs are functioning as designed.

I.1.D.3 NRC Criterion 3

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 alternate methodology is used, the application should be justified and applied to both venturi and the LEFM for comparison.

I.1.D.3.1 Response to NRC Criterion 3

Dominion uses a core thermal power uncertainty calculation approach consistent with ISA-RP67.04.02-2000, Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation (Reference I-14); and Cameron's Topical Report ER-80P (Reference I-1), as supplemented by ER-157P (Reference I-2). The combination of errors within instrument loops is accomplished per Dominion Technical Report EE-0116 (Reference I-15). This document is referenced in the Technical Specification Bases B 3.3.1. An alternate methodology for calculating UFM uncertainty was not used.

The fundamental approach used in the setpoint methodology is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then combined using the square root of the sum of the squares approach to determine the overall uncertainty. The same fundamental approach was used to determine the UFM based power calorimetric uncertainty. This approach has been approved by the NRC in Cameron Topical Reports ER-80P and ER-157P as well as for Seabrook Station Unit 1 (Reference I-11), Vogtle Electric Generating Plant (Reference I-12), and Cooper Nuclear Station (Reference I-13).

I.1.D.4 NRC Criterion 4

For plants where the LEFM was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), provide additional justification for use. This justification should show either that the meter installation is 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 the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated LEFM, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

I.1.D.4.1 Response to NRC Criterion 4

A UFM bounding uncertainty has been provided for use in the uncertainty calculation described below (References I-6 and I-7). The bounding calibration factor acceptability for the NAPS spool pieces was established by tests at the Alden Research Labs (References I-8 and I-9). These tests included a full-scale model of the Unit 1 and 2 hydraulic geometry and a straight pipe. An Alden Research Labs test data report and Cameron engineering report evaluating the test data were prepared. The calibration factor used for the UFM is based on these reports. The spool piece calibration factor uncertainty is based on the Cameron engineering reports. The site specific uncertainty analysis documents these analyses and will be maintained as part of the NAPS technical basis for the power uprate.

Final site-specific uncertainty analyses acceptance will occur after completion of the commissioning process. The commissioning process verifies bounding calibration test data and provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. Final commissioning is expected to be completed by March 2010.

I.1.E Total Power Measurement Uncertainty at North Anna Units 1 and 2

The overall thermal power uncertainty using the UFM is 0.35% at RTP. The uncertainty calculations for North Anna Units 1 and 2 are documented in References I-6 and I-7, which are Cameron proprietary documents that will be transmitted to the NRC via separate proprietary letter from Dominion. The key parameters and their uncertainty are summarized in Table I-1. In addition to the calorimetric inputs provided by the UFM for determination of feedwater mass flow rate and enthalpy, the North Anna plant computer uses several process inputs (e.g., charging flow, letdown flow, steam generator blowdown flow) to calculate the contribution of steam enthalpy and other gains and losses that are identified as Items 7 and 8 in Table I-1. For comparison, baseline values from Cameron ER-157P, Revision 5 (Reference I-2) are presented in Table I-1. Differences between the North Anna uncertainties and those from ER-157P, Revision 5 are a result of plant-specific calculations and parameter uncertainties.

The uncertainty for transducer installation, as identified in Cameron Customer Information Bulletin CIB-125 (Reference I-10), has been included in the UFM uncertainty for North Anna Unit 1 (Reference I-6) and North Anna Unit 2 (Reference I-7). These system uncertainties incorporate an additional transducer variability uncertainty in both the profile factor uncertainty and in the installation uncertainty.

Table I-1
Total Thermal Power Uncertainty Determination for North Anna Units 1 and 2

Item	Parameter[1]	ER-157P, Rev. 5 Uncertainty	Unit 1 Uncertainty	Unit 2 Uncertainty
1	Hydraulics: Profile factor	0.25%	0.20%	0.19%
2	Geometry: Spool dimensions Spool piece alignment Spool piece thermal expansion	0.10%	0.15%	0.16%
3	Time Measurements Time of Flight Measurements Non-fluid delay	0.05%	0.15%	0.15%
4	Feedwater Density[2] [4] Feedwater Density/Correlation Feedwater Density/Temperature Feedwater Density/Pressure	0.07%	0.07%	0.07%
5	Subtotal: Mass Flow Uncertainty (Root Sum Square of Items 1-4)	0.28%	0.30%	0.30%
6	Feedwater Enthalpy[3] [4] Feedwater Enthalpy/Temperature Feedwater Enthalpy/Pressure Power Uncertainty, Thermal Expansion	0.08% 0.12%	0.09% 0.12%	0.09% 0.12%
7	Steam Enthalpy: Pressure input and moisture uncertainty	0.07%	0.05%	0.05%
8	Gains/Losses	0.07%	0.09%	0.09%
9	Total Thermal Power Uncertainty	0.33%	0.35%	0.35%

1. Items 1 through 6 are directly associated with the UFM. Items 7 and 8 are based on other plant process inputs.
2. Density errors due to the density correlation, the UFM feedwater temperature determination and the feedwater pressure measurement.
3. Enthalpy errors due to the enthalpy correlation, the UFM feedwater temperature determination and the feedwater pressure measurement.
4. The bounding uncertainties in pressure and temperature are +15 psi and +0.57°F, respectively.

I.1.F Calibration and Maintenance Procedures of Instruments Affecting the Power Calorimetric

Information to specifically address the following aspects of the calibration and maintenance procedures related to the instruments that affect the power calorimetric.

I.1.F.i Maintaining Calibration

Calibration and maintenance for the UFM hardware and instrumentation will be performed using procedures based on the appropriate Cameron LEFM CheckPlus technical manuals, which ensures that the UFM remains bounded by the Topical Report ER-80P analysis and assumptions. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work is planned and executed in accordance with established NAPS work control processes and procedures. Routine preventive maintenance activities for the UFM will include, but not be limited to, those activities specified in Section I.1.D.1.1.

I.1.F.ii Controlling Software and Hardware Configuration

The LEFM CheckPlus system is designed and manufactured per Cameron's 10 CFR 50, Appendix B, Quality Assurance Program and Verification and Validation (V&V) Program. Cameron's V&V Program fulfills the requirements of ANSI/IEEE-ANS Standard 7-4.3.2, 1993 (Reference I-16) and ASME NQA-2a-1990 (Reference I-17). After installation, the UFM software configuration will be maintained using existing procedures and processes, which include verification and validation of software configuration changes. UFM hardware and the calorimetric process instrumentation will be maintained per the NAPS configuration control processes.

I.1.F.iii Performing Corrective Actions

Plant instrumentation that affects the power calorimetric, including the UFM inputs, will be monitored by NAPS personnel. Problems detected are documented per the NAPS corrective action process and necessary actions are planned and implemented.

I.1.F.iv Reporting Deficiencies to the Manufacturer

Conditions found to be adverse to quality will be documented per the NAPS corrective action program and reported to the vendor as needed to support corrective action.

I.1.F.v Receiving and Addressing Manufacturer Deficiency Reports

NAPS has existing processes for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the NAPS corrective action program and actions will be controlled by the NAPS work control process.

I.1.G Completion Time and Technical Basis

A completion time of 48 hours is proposed for operation at any power level in excess of 2893 MWt with the UFM not functional, provided that steady-state conditions persist throughout the 48-hour period. The basis for the proposed 48 hours completion time follows.

- Operations procedures will direct the use of the back-up calorimetric in the event of UFM failure. This algorithm receives input from alternate plant instruments (feedwater venturis and RTDs) for feedwater flow rate calculation. The feedwater flow from the three venturis will be normalized to the UFM feedwater flow rate, so that the alternate calorimetric matches the primary UFM based calorimetric. Also, the feedwater temperature and feedwater pressure measurements will be normalized to the more accurate data from the UFM. Alternate instrumentation accuracy due to nozzle fouling or transmitter drift will not significantly change over 48 hours. The feedwater flow venturis were inspected in fall 2007 (Unit 1) and fall 2008 (Unit 2) during recalibration. No venturi fouling was identified. This was the first visual inspection since plant startup. Based on the recent inspection results, it is very unlikely that venturi fouling or defouling would occur during the 48-hour completion time.
- UFM repairs are expected to be completed within an 8-hour shift. A completion time of 48 hours provides plant personnel sufficient time to plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.
- The normalized calorimetric instrumentation retains the accuracy of the UFM above 90% RTP. However, if the plant experiences a power decrease below 2893 MWt (98.4% of the uprate RTP) during the 48 hour period, the maximum permitted power level will be the current licensed core power level of 2893 MWt. This simplifies the Technical Requirements Manual (TRM) statement for Applicability, Condition, Required Action and Completion Time. Further, it is conservative to limit the power level to $\leq$ 98.4% RTP until the UFM is returned to functional status.
- As described in Cameron report ER-157P (Reference I-2), the UFM consists of two planes of transducers. Although a single path or single plane malfunction results in a minimal increase in feedwater flow uncertainty, operators will conservatively respond to a single path or single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode.

The 48-hour completion time will begin at the time the failure is annunciated in the main control room. A control room annunciator response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine CheckPlus System status and cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operator and maintenance response actions.

I.1.H Actions for Exceeding Completion Time and Technical Basis

The UFM functionality requirements will be contained in the NAPS Technical Requirements Manual (TRM). The limiting condition for operation for the TRM states that a functional UFM shall be used to perform the daily calorimetric heat balance measurements. If the UFM is declared not functional, the limiting condition for operation will require that either the UFM is restored to functional status within 48 hours or power is reduced to ≤ 2893 MWt.

In the event the UFM is not functional, the feedwater flow rate and feedwater temperature inputs to the calorimetric will be determined by alternate instrumentation. The existing feedwater venturi flow nozzles and RTDs will be used for the calorimetric until the UFM is returned to functional status. To ensure that the venturi based calorimetric is consistent with the UFM based calorimetric, the venturi based flow rate, feedwater temperature, and feedwater pressure will be normalized to the UFM. A plant computer loss is treated as a loss of both the UFM and the ability to obtain corrected calorimetric power using the alternate plant instrumentation. Operation at the uprated power level may continue until the next required nuclear instrumentation heat balance, which could be up to 24 hours. The plant computer failure will require reducing core thermal power to ≤ 2893 MWt as needed to support a manual calorimetric power calculation. These requirements ensure that a functional low uncertainty input is used whenever core power is greater than 2893 MWt. The operators will be provided with procedural guidance for those occasions when the UFM is not functional.

I REFERENCES

I-1 Cameron Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System*, Caldon Inc., March 1997.

I-2 Cameron Engineering Report ER-157P, Revision 5, *Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System*, Caldon Inc., October 2001.

I-3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry, TU Electric, Comanche Peak Steam Electric Station, Units 1 and 2 – *Review of Caldon Engineering Topical Report ER 80P, Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System* (ADAMS accession number 9903190065, legacy library), March 8, 1999.

I-4 Letter from S.A. Richards, NRC, to M.A. Krupa, *Entergy, Waterford Steam Electric Station, Unit 3; Riverbend Station; and Grand Gulf Nuclear Station – Review of Caldon Engineering Report ER-157P* (ML013540256), December 20, 2001.

I-5 NRC Letter, B.E. Thomas to E.M. Hauser, *Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus Ultrasonic Flow Meters (UFM)* (ML061700222), July 5, 2006.

I-6 Cameron Caldon® Ultrasonics Engineering Report ER-646, Revision 2, *Bounding Uncertainty Analysis for Thermal Power Determination at North Anna Unit 1 Using the LEFM CheckPlus System*, January 2009.

I-7 Cameron Caldon® Ultrasonics Engineering Report ER-637, Revision 3, *Bounding Uncertainty Analysis for Thermal Power Determination at North Anna Unit 2 Using the LEFM CheckPlus System*, January 2009.

I-8 Cameron Caldon® Ultrasonics Engineering Report 689, Rev. 2, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for North Anna Unit 1*, January 2009.

I-9 Cameron Caldon® Ultrasonics Engineering Report 675, Rev. 2, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for North Anna Unit 2*, January 2009.

I-10 *Cameron Customer Information Bulletin CIB-125*, Revision 0, April 23, 2007.

I-11 NRC letter to FPL Energy Seabrook, *Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML061360034, May 22, 2006.

I-12 NRC letter to Vogtle Electric Generating Plant, Units 1 and 2, *Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate*, ML080350347, February 27, 2008.

I-13 NRC letter to Nebraska Public Power District, Cooper Nuclear Station, *Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML081540280, June 30, 2008.

I-14 ISA-RP67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation*.

I-15 Dominion Technical Report EE-0116, Revision 4, *Allowable Values for North Anna Improved Technical Specifications (ITS) Tables 3.3.1-1 and 3.3.2-1 and Setting Limits for Surry Custom Technical Specifications (CTS)*, Sections 2.3 and 3.7, September 30, 2008.

I-16 ANSI/IEEE-ANS Standard 7-4.3.2, 1993, *IEEE Standard Criteria for Digital Computers in Safety System of Nuclear Power Generating Station.*

I-17 ASME NQA-2a-1990, *Quality Assurance Requirement for Nuclear Facility Application.*

II ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

   A. Identify the transient or accident that is the subject of the analysis

   B. Confirm and explicitly state that

      i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

      ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

      iii. the analyses of record are not changed by the requested power uprate

   C. Confirm that bounding event determinations continue to be valid

   D. Provide a reference to the NRC's previous approvals discussed in Item B. above

RESPONSE TO II - ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

II. Accidents and Transients Bounded by the Analyses of Record for the Measurement Uncertainty Recapture

II.1 Introduction

A review of UFSAR Chapters 6 and 15 and other related subsections was performed to support the NAPS MUR power uprate with respect to the accident analyses. Evaluations were also performed on other analyses (e.g., internal flooding, station blackout, ATWS). The UFSAR review was conducted to confirm that the existing analyses of record, as currently presented in the UFSAR, were performed conservatively and remain valid and bounding for the proposed power uprate. Table II-1 indicates the analysis power levels used for the NAPS MUR power uprate.

Table II-1
Analysis Power Levels for North Anna Units 1 and 2 MUR Uprating

Analysis Scope	Core Power MWt	NSSS Power MWt[(4)]	Source
NSSS	2956[(1)]	2968	Design Parameters
Safety Analyses	2951[(2)]	2963	UFSAR Chapters 6 and 15
Statistical DNBR Events	2942.2[(3)]	2955	UFSAR Chapter 15

1. 102% of current analyzed core power of 2898 MWt
2. 102% of current RTP of 2893 MWt
3. 101.7% of current RTP of 2893 MWt
4. The analyses use 12 MWt for RCP net heat addition.

The analyses generally model the core and/or NSSS thermal power in one of three ways. First, some analyses apply a 2.0% increase to the initial power level to account for the power measurement uncertainty. These analyses have not been re-performed for the MUR uprate conditions, because the sum of the proposed core power level and the decreased power measurement uncertainty falls within the previously analyzed conditions. The existing 2.0% uncertainty is reallocated so a portion is applied to uprate power and the remainder is retained to accommodate the power measurement uncertainty. Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the proposed power level. Third, some of the analyses are performed at 0% power conditions or do not actually model core power level. These analyses have not been re-performed because they are unaffected by the core power level.

For the NAPS MUR power uprate, a core RTP of 2940 MWt was selected based on the calorimetric uncertainty of 0.35% with the UFM and a review of the accident analysis assumptions for core power. The deterministic accident analyses use 2951 MWt (102% of 2893 MWt) as the total core power, which leaves 11 MWt of margin to accommodate the power uncertainty. The 11 MWt is 0.37% of 2940 MWt. Since the power calorimetric uncertainty of 0.35% at RTP with the UFM is less than the accident analysis allowance of 0.37% with a 2940 MWt licensed power level, the deterministic accident analyses are bounding for the MUR power uprate. The statistical DNBR events were analyzed previously at 2942.2 MWt and remain bounding for the proposed RTP of 2940 MWt. In conclusion, the evaluations of the UFSAR events in Section II.2 support an uprated RTP of 2940 MWt.

II.1.A North Anna Fuel

NAPS has transitioned from Westinghouse North Anna Improved Fuel (NAIF) to AREVA Advanced Mark-BW (AMBW) fuel. NAPS UFSAR Chapters 4 and 15 describe the features and analyses for both Westinghouse NAIF and AREVA AMBW fuel. Westinghouse NAIF fuel is not currently used in North Anna core designs. Thus, the NAIF fuel analyses were not evaluated for the MUR power uprate; fuel evaluations were performed only on the AREVA AMBW fuel. If there is a future need to re-use irradiated Westinghouse NAIF fuel assemblies in a North Anna core, peaking factors for Westinghouse re-use fuel would be significantly reduced from the peaking factor limits used in UFSAR Chapters 4 and 15. Dominion has an NRC approved reload analysis methodology (Reference II-1) that would be used to evaluate the NAIF fuel and define fuel-specific peaking factor limits for the cycle-specific Core Operating Limits Report (COLR).

II.1.B DNBR Analysis of AREVA Advanced Mark-BW Fuel

The UFSAR Chapter 15 DNBR analyses for the AREVA AMBW fuel product use the NRC approved VIPRE-D/BWU code/correlation pair. The NRC approved Dominion's license amendment request (Reference II-2) for implementation of VIPRE-D, using the Statistical DNBR Evaluation Methodology from Dominion Topical Report VEP-NE-2-A in Reference II-4. The DNBR analyses described in Reference II-2 were based on 101.7% of the current licensed power level of 2893 MWt. The UFSAR Condition II events and applicable Condition III and IV events were either evaluated or explicitly analyzed for DNBR acceptance as part of the VIPRE-D/BWU implementation. The Dominion VIPRE-D analyses used the same 101.7% uprated power statepoints that were developed for the AREVA AMBW fuel transition (Reference II-17). Table II-2 reflects the DNBR analysis basis of 101.7% of 2893 MWt. The VIPRE-D/BWU code/correlation limits used a core power uncertainty of ± 2.2% two sigma (2σ) standard deviation. Although the power calorimetric uncertainty will decrease with the use of ultrasonic flow meters, the statistical DNBR limit will continue to be based on ± 2.2% for conservatism.

II.1.C Accident/Transient/Other Analyses Matrix

Table II-2 below provides a brief overview of the accident/transient analyses and other analyses contained in the NAPS UFSAR (Reference II-3), the assumed core power level in each analysis, and whether these analyses remain bounding for the MUR power uprate. This table also provides references to the NRC's previous approval of each analysis, if applicable, or a statement confirming that NRC approved methods were used in the analysis of record that was implemented under the provisions of 10 CFR 50.59. A discussion of each UFSAR event is presented in Section II.2, Discussion of Events.

Table II-2
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	15.2.1	0	Yes	NRC approval in Reference II-4.
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	15.2.2	102(1)	Yes	NRC approval in Reference II-4 for DNBR. Overpressure analyses were performed using NRC approved methodologies.
Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)	15.2.3	101.7	Yes	NRC approval in Reference II-4.
Uncontrolled Boron Dilution	15.2.4	102(1)	Yes	NRC approval in Reference II-4 for DNBR. Overpressure analyses were performed using NRC approved methodologies.
Partial Loss of Forced Reactor Coolant Flow	15.2.5	101.7	Yes	Partial Loss of Flow is bounded by the Complete Loss of Forced Reactor Coolant Flow event. NRC approval in Reference II-4.
Start-up of an Inactive Reactor Coolant Loop	15.2.6	N/A	N/A	Technical Specification 3.4.4 (Reactor Coolant System) prohibits power operation with less than three reactor coolant loops in service.
Loss of External Electrical Load and/or Turbine Trip	15.2.7	102(1)	Yes	NRC approval in Reference II-4 for DNBR. Overpressure analyses were performed using NRC approved methodologies.
Loss of Normal Feedwater	15.2.8	102	Yes	Analysis was performed using NRC approved methodologies.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval
Loss of Offsite Power to the Station Auxiliaries	15.2.9	102	Yes	Analysis was performed using NRC approved methodologies.
Excessive Heat Removal Due to Feedwater System Malfunction	15.2.10	101.7	Yes	NRC approval in Reference II-4.
Excessive Load Increase Incident	15.2.11	101.7	Yes	NRC approval in Reference II-4.
Accidental Depressurization of the Reactor Coolant System	15.2.12	101.7	Yes	NRC approval in Reference II-4.
Accidental Depressurization of the Main Steam System	15.2.13	0	Yes	NRC approval in Reference II-4.
Spurious Operation of the Safety Injection System at Power	15.2.14	102[(1)]	Yes	NRC approval in Reference II-4 for DNBR. RCS overpressure analysis was performed using NRC approved methodologies.
Small Break Loss of Coolant Accident	15.3.1.8 – 15.3.1.14	102	Yes	NRC approval of the small break loss of coolant accident (SBLOCA) peak clad temperature (PCT) analysis in References II-5 and II-6. The analysis has been supplemented by additional evaluations under the provisions of 10 CFR 50.46.
Minor Secondary System Pipe Breaks	15.3.2	0	N/A	Minor Secondary System Pipe Breaks are bounded by the main steam line break (MSLB) accident. Analysis was performed using NRC approved methodologies.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval
Inadvertent Loading of a Fuel Assembly Into an Improper Position	15.3.3	N/A	N/A	N/A
Complete Loss of Forced Reactor Coolant Flow	15.3.4	101.7	Yes	NRC approval in Reference II-4.
Waste Gas Decay Tank Rupture	15.3.5	N/A(2)	Yes	NRC approval in Reference II-10
Volume Control Tank Rupture	15.3.6	N/A(2)	Yes	(5)
Single Rod Cluster Control Assembly Withdrawal at Power	15.3.7	102(1)	Yes	NRC approval in Reference II-4 for DNBR. Overpressure analyses were performed using NRC approved methodologies.
Breaks in Instrument Lines or Lines From Reactor Coolant System That Penetrate Containment	15.3.8	N/A	N/A	N/A
Large Break Loss of Coolant Accident	15.4.1.10-16	100	No	NRC approval in References II-5 and II-6.
Large Break Loss of Coolant Accident (long-term cooling)	15.4.1.17	102	Yes	NRC approval of analyses for post-LOCA containment sump boron concentration, containment sump pH, and hot leg switchover time in Reference II-7 and supplemented by additional evaluations under the provisions of 10 CFR 50.59.
Major Secondary System Pipe Rupture (Main Steam Line Break)	15.4.2.1	0	Yes	NRC approval in Reference II-4.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval
Major Secondary System Pipe Rupture (Main Feed Line Break)	15.4.2.2	102	Yes	Analysis was performed using NRC approved methodologies.
Steam Generator Tube Rupture	15.4.3	102	Yes	Analysis was performed using NRC approved methodologies.
Locked Reactor Coolant Pump Rotor	15.4.4	102(1)	Yes	NRC approval in Reference II-4 for DNBR. Overpressure analyses were performed using NRC approved methodologies.
Fuel Handling Accident (FHA)	15.4.5	102.2	Yes	NRC approval in Reference II-24.
Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6	102	Yes	NRC approval in References II-5, II-6 and II-8.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Other Analyses	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval
Natural Circulation Cooldown	15.2.9	102[6]	Yes	Refer to UFSAR Section 15.2.9 (Loss of Offsite Power) above and Section 9.5.1 (Safe Shutdown Fire Analysis – Appendix R) below for NRC approval references.
Long-term LOCA Mass and Energy Release	6.2.1.1.1.3	102	Yes	NRC approval in Reference II-9.
Short-term LOCA Mass and Energy Release	6.2.1.1.2	102.2[7]	Yes	NRC approval in Reference II-10.
Main Steam Line Break Mass and Energy Release	6.2.1.3.1.2.1	102.2	Yes	NRC approval in References II-9 and II-11.
ATWS/AMSAC	4.3.1.7/ 7.7.1.14	102	Yes	NRC approval in Reference II-12.
Station Blackout	8.1.2[3]	102	Yes	NRC approval in References II-13, II-14 and II-15.
Analyses to Determine EQ Parameters	3.11	100.24	No	NRC approval in References II-29 and II-30.
Safe Shutdown Fire Analysis (Appendix R report)	9.5.1[4]	102	Yes	NRC approval in References II-27 and II-28.
Spent Fuel Pool Cooling	9.1.3	102	Yes	NRC approval in Reference II-10.
Internal Flooding	2.4.10	N/A	Yes	NRC approval in Reference II-31

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Other Analyses	UFSAR Section	Assumed Reactor Power Level (% of 2893 MWt)	Bounding (Yes/No)	NRC Approval

1. Events that have DNBR plus other non-DNBR analyses (e.g., RCS/Main Steam system overpressure) assume a core power level of 101.7% of 2893 MWt for the DNBR analyses and 102% of 2893 MWt for the non-DNBR analyses. Refer to Section II.1.B for a detailed discussion of assumed core power levels
2. Based on 1% failed fuel fission product inventory in the RCS
3. The station blackout analysis is not described in UFSAR Section 8.1.2, where the station blackout equipment is defined. The analyses are maintained in engineering calculations.
4. The Appendix R safe shutdown analyses are not described in UFSAR Section 9.5.1, where the fire protection equipment is specified. The post-fire safe shutdown analyses are maintained in engineering calculations
5. The original VCT rupture analysis was submitted in the FSAR, which was NRC approved by Reference II-10. However, the NRC did not specifically discuss the VCT rupture analysis in Reference II-10. The original analysis has been subsequently updated under the provisions of 10 CFR 50.59.
6. The assumed power level only applies to the Loss of Offsite Power analysis.
7. The short-term LOCA mass and energy releases are affected by changes in RCS temperatures, which are a function of core power. Evaluations confirmed that the UFSAR analyses for LOCA mass and energy releases at 102.2% of 2893 MWt used conservative RCS temperatures compared to design RCS temperatures for MUR power uprate.

II.2 Discussion of Events

UFSAR Chapter 15 accidents/transients and other UFSAR analyses were reviewed to support the NAPS MUR power uprate. A summary of each evaluation is provided below.

II.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition - UFSAR 15.2.1

The rod withdrawal from subcritical event is analyzed in UFSAR Section 15.2.1 for comparison to DNBR, RCS pressure, and main steam system (MSS) pressure limits. The RCS and MSS overpressure cases were performed using the NRC approved RETRAN methodology (Reference II-18). Because this event is evaluated at hot zero power conditions (0% rated core power), the UFSAR analyses of record are unaffected by the MUR power uprate. The analysis of record remains acceptable for the power uprate.

II.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power - UFSAR 15.2.2

The rod withdrawal at power event is analyzed in UFSAR Section 15.2.2 for comparison to DNBR, RCS pressure, and MSS pressure limits. The RCS and MSS overpressure cases were performed at 10%, 60%, 100%, and 102% of 2893 MWt using the NRC approved RETRAN methodology (Reference II-18). The most limiting case for overpressure occurred at 10% of 2893 MWt. The DNBR analysis at 101.7% of 2893 MWt was approved by the NRC with the implementation of VIPRE methodology for the AMBW fuel product (Reference II-4). The analyses are bounding for the MUR power uprate.

II.2.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error) - UFSAR 15.2.3

The rod control cluster assembly (RCCA) misalignment, dropped RCCA, and dropped RCCA bank events were analyzed to confirm that DNBR limits are met. RCCA misalignment was evaluated at 101.7% of 2893 MWt, with the minimum DNBR result remaining above the DNBR limit.

A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm, which will be detected by the power range negative flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the RCCA bank drop. The core is not adversely affected during this period due to the rapidly decreasing power. A single or multiple dropped RCCA is addressed by the dropped rod limit lines at 101.7% of 2893 MWt. Dropped rod limit lines are used to determine the allowable radial peaking factor at the limiting point during the dropped rod transient. The dropped RCCA event is evaluated each fuel cycle to ensure the dropped rod limit lines remain bounding for each core reload. The

RCCA (system malfunction or operator error) accident has been evaluated for the MUR power uprate and remains bounding.

II.2.4 Uncontrolled Boron Dilution - UFSAR 15.2.4

The uncontrolled boron dilution event is analyzed in UFSAR Section 15.2.4 for DNBR, RCS pressure, and MSS pressure. The at-power analyses for RCS and MSS overpressure are bounded by the rod withdrawal at power event, which introduces higher reactivity insertion rates and is thus bounding for uncontrolled boron dilution. DNBR analysis is bounded by the rod withdrawal at power event for at-power conditions and the rod withdrawal from subcritical event for startup conditions. Analysis is not required for subcritical conditions as uncontrolled boron dilution is precluded by locking out potential sources of primary grade water during these modes of operation. No specific DNB analysis was performed for this transient to support the AREVA AMBW fuel transition. Based on the acceptable rod withdrawal at power and rod withdrawal from subcritical DNB analyses, the uncontrolled boron dilution DNBR remains above the DNBR limit at 101.7% of 2893 MWt.

II.2.5 Partial Loss of Forced Reactor Coolant Flow - UFSAR 15.2.5

The partial loss of forced reactor coolant event is bounded by the complete loss of forced reactor coolant flow event (UFSAR Section 15.3.4) and is not explicitly analyzed.

II.2.6 Start-up of an Inactive Reactor Coolant Loop - UFSAR 15.2.6

The current start-up of an inactive loop design and licensing bases credit Technical Specification controls to preclude the possibility of a significant inadvertent reactivity addition during or following loop stop valve operations. Technical Specification 3.4.4 (Reactor Coolant System) prohibits power operation with less than three reactor coolant loops in service. The NAPS MUR power uprate review has not included any analyses that would provide the licensing basis for two loop power operation. The initial assumptions for FSAR analyses are that the plant is maintained within the limits of the Technical Specifications. Since two loop operation is and will remain prohibited by Technical Specifications, no analysis of this event is required for uprated power conditions. Because start-up of an inactive loop is a deliberate action under operator control governed by Technical Specifications, the sequence of operator errors required for a start-up of an inactive loop event to occur is considered non-credible. The start-up of an inactive loop event is therefore not affected by the MUR power uprate.

II.2.7 Loss of External Electrical Load and/or Turbine Trip - UFSAR 15.2.7

The loss of external electrical load and/or turbine trip event was evaluated for DNBR, and analyzed for MSS and RCS overpressure. This event was not explicitly analyzed for DNBR as part of the AREVA AMBW fuel transition.

However, DNBR for this event is bounded by acceptable analysis of the rod withdrawal at power event, core thermal limits, and axial offset envelopes at 101.7% of 2893 MWt.

The MSS and RCS overpressure cases were performed using the NRC approved RETRAN methodology (Reference II-18). Both the MSS and RCS overpressure cases used an initial reactor power of 102% of 2893 MWt. Therefore, the DNBR, RCS overpressure, and MSS overpressure cases for this event are bounding for the MUR power uprate.

II.2.8 Loss of Normal Feedwater - UFSAR 15.2.8

The loss of normal feedwater event was analyzed for RCS overpressure and pressurizer overfill leading to a loss of reactor coolant from the core. The analysis used the NRC approved RETRAN analysis methodology (Reference II-18) to determine the plant transient response. The analysis used an initial core power of 2951 MWt, or 102% of 2893 MWt. Therefore, the loss of normal feedwater analysis is bounding for the MUR power uprate.

II.2.9 Loss of Offsite Power to the Station Auxiliaries - UFSAR 15.2.9

The loss of offsite power was analyzed for RCS overpressure and pressurizer overfill. The analysis used the NRC approved RETRAN analysis methodology (Reference II-18) to determine the plant transient. The analysis used an initial core power of 2951 MWt, or 102% of 2893 MWt. Therefore, the analysis of record is bounding for the MUR power uprate.

II.2.10 Excessive Heat Removal Due to Feedwater System Malfunction - UFSAR 15.2.10

Excessive heat removal resulting from feedwater system malfunction events (excessive feedwater flow and feedwater temperature reduction) was evaluated for DNBR. No specific DNB analysis was performed for the excessive feedwater flow transient. Based on the acceptable DNB analysis of core thermal limits and axial offset envelopes at 101.7% of 2893 MWt, the DNBR for this case remains above the DNBR limit. This conclusion was extended to VIPRE-D implementation with the Dominion Statistical DNBR Evaluation Methodology (Reference II-4). The feedwater temperature reduction event has been evaluated for the MUR power uprate and is bounded by the excessive load increase event described in UFSAR Section 15.2.11. No explicit DNBR analysis was performed.

II.2.11 Excessive Load Increase Incident - UFSAR 15.2.11

The excessive load increase incident analysis is discussed in UFSAR Section 15.2.11 and was evaluated for DNBR. This analysis uses the NRC approved RETRAN analysis methodology (Reference II-18) to determine the plant transient response. No specific DNB analysis was performed for this transient.

Based on the acceptable DNB analysis of core thermal limits and axial offset envelopes at 101.7% of 2893 MWt, the DNBR for this case remains above the DNBR limit. This conclusion was extended to VIPRE-D implementation with the Dominion Statistical DNBR Evaluation Methodology (Reference II-4). Therefore, the excessive load increase event has been evaluated for the MUR power uprate.

II.2.12 Accidental Depressurization of the Reactor Coolant System - UFSAR 15.2.12

The accidental depressurization of the RCS analysis uses the NRC approved RETRAN analysis methodology (Reference II-18) and was evaluated for DNBR. No specific DNB analysis was performed for this transient. Based on the acceptable DNB analysis of core thermal limits and axial offset envelopes at 101.7% of 2893 MWt, the DNBR for this case remains above the DNBR limit. This conclusion was extended to VIPRE-D implementation with the Dominion Statistical DNBR Evaluation Methodology (Reference II-4). Therefore, the accidental depressurization of the RCS event has been evaluated for the MUR power uprate.

II.2.13 Accidental Depressurization of the Main Steam System - UFSAR 15.2.13

The accidental depressurization of the MSS analysis uses the NRC approved RETRAN analysis methodology (Reference II-18) to perform the transient response, and was evaluated for DNBR. The analysis is performed at zero power. The statepoint from this calculation was evaluated for the AREVA AMBW fuel transition and for the MUR power uprate. No specific DNB analysis was performed for this transient because it was bounded by the MSLB statepoint. Since this accident is limiting at 0% power, the MSS depressurization analysis of record is unaffected by the MUR power uprate.

UFSAR Section 15.2.13.3 describes the results of a decay heat release piping break. The analysis at 0% power was more limiting than an analysis performed at 102% of 2893 MWt. Since the limiting case is analyzed at 0% power, the analysis of record is unaffected by the MUR power uprate.

II.2.14 Spurious Operation of the Safety Injection System at Power - UFSAR 15.2.14

The spurious operation of the safety injection system at power is evaluated to demonstrate that DNBR limits are met, that RCS pressure remains less than 110% of the design limit, and that the event does not propagate into a SBLOCA (ANS Condition III Event).

The DNBR analysis of record was prepared for the core power uprate submittal in 1985 (Reference II-19 Section 3.1.3.3.10). The full-power DNBR analysis used a nominal core power of 2898 MWt. The event was evaluated for the AREVA AMBW fuel transition and the MUR power uprate. No specific DNB analysis was performed for this transient, because the transient DNBR increases above the initial value and remains above the DNBR limit. This conclusion was extended to VIPRE-D implementation with the Dominion Statistical DNBR Evaluation Methodology (Reference II-4).

The analysis of record includes an evaluation of potential RCS pressurization as a result of a spurious safety injection. The evaluation concluded that a single pressurizer safety valve provides adequate relief capacity assuming a spurious safety injection and concurrent post-trip RCS heatup. The heatup was assumed to progress from a no-load nominal average temperature and driven by decay heat, while assuming no secondary heat sink. The decay heat was conservative for a core power of 102% of 2893 MWt. Therefore, the analysis of record is bounding for the MUR power uprate.

The spurious safety injection was evaluated to assess its potential to propagate into a SBLOCA event if one or more of the pressurizer safety valves or PORVs were to fail open and isolation was not possible. Pressurizer safety valves and PORVs are qualified for liquid relief, and testing has resulted in no instances of failure to reseat following liquid relief. The resulting leakage is within the normal makeup system capacity and is not considered a SBLOCA event. Therefore, a spurious safety injection completely filling the pressurizer with water relief through a safety valve does not constitute a failure to meet the event propagation acceptance criterion. No specific analysis assumption is made for the time to terminate a spurious safety injection. This basis is independent of core power level.

The spurious safety injection event has been evaluated for the proposed MUR power uprate and applicable acceptance criteria continue to be met.

II.2.15 Loss of Coolant From Small Ruptured Pipes or From Cracks in Large Pipes That Actuates the Emergency Core Cooling System (Small Break Loss of Coolant Accident) - UFSAR 15.3.1

UFSAR Sections 15.3.1.8 through 15.3.1.14 describe the SBLOCA analyses for AREVA AMBW fuel. The SBLOCA analyses were approved by the NRC with the implementation of the AMBW fuel product in Reference II-5 for Unit 1 and in Reference II-6 for Unit 2. The approved SBLOCA analyses have been augmented by evaluations under 10 CFR 50.46. The SBLOCA analyses assume a core power of 2951 MWt, or 102% of 2893 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.16 Minor Secondary System Pipe Breaks - UFSAR 15.3.2

Minor secondary system pipe breaks must be accommodated with only a small fraction of fuel element failures. Since the major secondary system pipe rupture analysis meets this criterion, it bounds the minor secondary system pipe break and a separate analysis is not required.

The MSLB analyses demonstrate that the consequences of a minor secondary pipe break are acceptable, because a DNBR less than the limit does not occur even for a more critical limiting major secondary system pipe break. Since the MSLB accident is limiting at 0% power, the minor secondary system pipe break is unaffected by the MUR power uprate.

II.2.17 Inadvertent Loading of a Fuel Assembly into an Improper Position - UFSAR 15.3.3

This event is described in UFSAR Section 15.3.3. The incore movable flux detectors are used to verify power shapes at the beginning of each cycle. The system is capable of detecting a fuel assembly enrichment error or loading error that causes power shape peaking in excess of the design value. Detailed power distribution measurements are conducted at defined power levels during the power ascension following a refueling outage. The power distributions are compared to predicted values. In the unlikely event that a loading error occurs, the resulting power distribution effects would either be detected by the incore moveable detector system, or cause a power perturbation small enough to fall within the allowable uncertainties between nominal and design power shapes. These same power distribution validations would identify fuel and core loading errors at the MUR uprated power level.

The analyses are from the original FSAR showing representative results of core mis-loading errors to demonstrate the high likelihood of detecting these errors with the incore flux mapping program. The core mis-loading scenario is not analyzed for each core reload. The event conclusions are not impacted by the MUR power uprate.

II.2.18 Complete Loss of Forced Reactor Coolant Flow - UFSAR 15.3.4

UFSAR Section 15.3.4 describes the complete loss of forced reactor coolant flow accident, which was analyzed for DNBR. The transient analysis of the complete loss of forced reactor coolant flow event using the NRC approved RETRAN methodology (Reference II-18) was last performed as part of the transition to AREVA AMBW fuel. The NRC reviewed the analysis with the license amendment allowing transition to AREVA AMBW fuel at NAPS (References II-5 and II-6). The DNBR statepoint analysis for the complete loss of forced reactor coolant flow was last reviewed by the NRC with the approval for use of the VIPRE-D thermal hydraulic code (Reference II-4).

The complete loss of forced reactor coolant flow analysis considers the effects of a 1.7% power uprate by assuming a nominal initial core power level of 2942.2 MWt (2893 MWt × 1.017). The detailed core thermal-hydraulic analysis models the AREVA AMBW fuel product using the Virginia Power Statistical DNBR Evaluation Methodology with the VIPRE-D thermal-hydraulic computer code for both the underfrequency and undervoltage cases. The analysis produced a minimum DNBR above the DNBR limit. Thus, the complete loss of forced reactor coolant flow event has been analyzed previously for a 1.7% MUR power uprate.

II.2.19 Waste Gas Decay Tank Rupture - UFSAR 15.3.5

The waste gas decay tank rupture analysis was part of the original plant licensing basis. The analysis resulted in an exclusion area boundary whole body dose of 1.6 rem, which is reported in UFSAR Section 15.3.5 and compared to the 10 CFR 100 acceptance criterion. The 10 CFR 100 acceptance criterion for waste gas decay tank rupture exclusion area boundary whole body dose was 25 rem. Conservatisms in the radiological atmospheric dispersion factor (X/Q), dose conversion factors, and gap activities that were used in the analysis are such that the MUR power uprate impact on the waste gas decay tank rupture accident consequences is bounded.

II.2.20 Volume Control Tank Rupture - UFSAR 15.3.6

The VCT rupture dose analysis is described in UFSAR Section 15.3.6.2 and is based on the guidance from NRC Branch Technical Position 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 this analysis, the MUR power uprate impact on the VCT rupture radiological consequences is bounded.

II.2.21 Single Rod Cluster Control Assembly Withdrawal at Power - UFSAR 15.3.7

DNBR for the single RCCA withdrawal at power event is bounded by the analysis of uncontrolled RCCA bank withdrawal at power event, core thermal limits, and axial offset envelopes at 101.7% of 2893 MWt, which remain above the DNBR limit. The single RCCA withdrawal at power event was not explicitly analyzed for DNBR as part of the AREVA AMBW fuel transition. The NRC approved the single RCCA withdrawal at power event DNBR analysis in Reference II-4.

The uncontrolled RCCA bank withdrawal at power event bounds the RCS and MSS overpressurization from a single RCCA withdrawal. Therefore, no specific RCS or MSS overpressurization analysis was performed for a single RCCA withdrawal at power. Pressurizer overfill is not a concern for this event. The MUR power uprate is bounded by the analysis of record.

II.2.22 Breaks in Instrument Lines or Lines From Reactor Coolant System That Penetrate Containment - UFSAR 15.3.8

There are no instrument lines penetration the containment that contain reactor coolant. Therefore, this event has no explicit analysis and is unaffected by the MUR power uprate.

II.2.23 Loss of Reactor Coolant From Ruptured Pipes or From Cracks in Large Pipes Including Double Ended Rupture That Actuates the Emergency Core Cooling System (Large Break Loss of Coolant Accident) - UFSAR 15.4.1

UFSAR Section 15.4.1.17 summarizes the large break LOCA long-term cooling evaluations for initial ECCS operation, considers the long-term water supply to the core, and discusses the procedures to mitigate boric acid build-up in the core. This UFSAR section reiterates the results of LOCA analyses performed in other UFSAR sections. The power uprate required an AREVA AMBW fuel large break LOCA analysis for peak clad temperature and fuel oxidation (UFSAR Sections 15.4.1.10 through 15.4.1.17). This analysis is described in Section III. Other post-LOCA analyses that demonstrate long-term core cooling are described below.

The analyses of record for post-LOCA containment sump boron concentration (subcriticality), containment sump pH, and hot leg switchover time were reviewed for the MUR power uprate. The most recent NRC approval (Reference II-7) of these analyses was in a license amendment that increased the boron concentration limits in the RWST, casing cooling tank, accumulators, and spent fuel pool during refueling. These analyses have been supplemented by additional evaluations performed under the provisions of 10 CFR 50.59. The following evaluations confirm that the analyses of record remain bounding for the proposed power uprate, and that long-term core cooling is assured.

- The containment sump pH calculation does not explicitly include a core power level. The methodology normalizes the contributing inventories to a sump temperature of 70°F. The proposed core power increase does not affect the analysis that determines the post-LOCA sump pH.
- The minimum containment sump boron concentration calculation to ensure post-LOCA subcriticality does not explicitly include a core power level. Each core reload confirms that the post-LOCA sump boron concentration provides adequate subcriticality during the vessel reflood stage, during the switchover to cold leg recirculation, and during long-term core cooling. Core power level is accounted for in the core reload confirmation of post-LOCA sump boron concentration limits.
- The hot leg switchover time calculation uses a core power level of 2951 MWt, or 102% of 2893 MWt, to determine the post-LOCA core steaming rate. This analysis remains bounding for the proposed MUR power uprate.

II.2.24 Major Secondary System Pipe Rupture - UFSAR 15.4.2

Major secondary system pipe rupture includes both main steam line rupture and main feedwater pipe rupture.

The MSLB is discussed in UFSAR Section 15.4.2.1 and was analyzed for DNBR and containment integrity. The MSLB accident analysis for DNBR is limiting at 0% power, so the MSLB analysis is unaffected by the MUR power uprate.

UFSAR Section 15.4.2.2 describes the main feedwater line break accident. This event is evaluated for RCS overpressurization and core integrity, but it is not a DNB limiting event. The RCS overpressurization analysis was performed using the NRC approved RETRAN methodology from an initial power level of 2951 MWt, or 102% of 2893 MWt. The main feedwater line break accident analysis is bounding for the MUR power uprate.

II.2.25 Steam Generator Tube Rupture - UFSAR 15.4.3

The steam generator tube rupture (SGTR) accident is discussed in UFSAR Section 15.4.3. The accident analyses demonstrate that the radiological dose consequences are less than the regulatory limits and that SG overfill does not occur.

The thermal-hydraulic analysis of record uses the NRC approved RETRAN analysis methodology (Reference II-18) to predict the ruptured SG break flow and the RCS and secondary system response. RETRAN also calculates the fraction of the break flow that flashes directly to steam for input to the dose analysis, and steam releases from the ruptured and intact SGs through the main steam safety valves (MSSVs) and SG PORVs. The analysis assumed a core power of 2951 MWt, or 102% of 2893 MWt, to generate the steam release rates. Therefore, the analyzed core power is bounding for the MUR power uprate.

No explicit safety analysis is performed to demonstrate that liquid inventory does not enter the main steam lines (SG overfill). The basis for not having an explicit analysis is industry experience during actual SGTR events (Ginna, North Anna, Surry, Prairie Island) and simulator exercises that validated the emergency operating procedures. The small increase in core power will not reduce the emergency operating procedures effectiveness in preventing SG overfill. Therefore, explicit deterministic analyses to address SG overfill are not performed. The SGTR accident analysis is bounding for the MUR power uprate.

II.2.26 Locked Reactor Coolant Pump Rotor - UFSAR 15.4.4

The RCP locked rotor/sheared shaft events are described in UFSAR Section 15.4.4, and are analyzed for DNBR, and RCS and MSS overpressure. The transient analysis of the RCP locked rotor/sheared shaft event using the NRC approved RETRAN methodology (Reference II-18) was last performed as part of the transition to AREVA AMBW fuel. The NRC reviewed the analysis with the license amendment allowing transition to AREVA AMBW fuel at NAPS (References II-5 and II-6). The DNBR statepoint analysis for the RCP locked rotor/sheared shaft event was last reviewed by the NRC with the approval for use of the VIPRE-D thermal-hydraulic code (Reference II-4).

The RCP locked rotor/sheared shaft analysis considers the effects of a 1.7% power uprate by assuming a nominal initial core power level of 2942.2 MWt (2893 MWt $\times$ 1.017). The detailed core thermal-hydraulic analysis for the AREVA AMBW fuel product is performed using the Virginia Power Statistical DNBR Evaluation Methodology with the VIPRE-D thermal-hydraulic computer code. The most limiting statepoint for RCP locked rotor/sheared shaft was identified and evaluated at 101.7% of 2893 MWt.

RCS and MSS overpressure cases were analyzed at 102% of 2893 MWt, with results of the RCP locked rotor/sheared shaft event leading to an RCS and MSS pressure less than the design limit. Thus, the RCP locked rotor/sheared shaft event has been analyzed previously for a 1.7% MUR power uprate.

II.2.27 Fuel Handling Accident – UFSAR 15.4.5

The current fuel handling accident radiological analysis is based upon the alternate source term (AST) as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current fuel handling accident analysis is a function of core power, enrichment, burnup, gap fractions for non-LOCA events from Regulatory Guide 1.183, the number of failed fuel rods, and the assumed radial peaking factor. The existing fuel handling accident dose evaluation was performed using a core inventory that assumes 2958 MWt, which is 102.2% of 2893 MWt, and a single failed fuel assembly (264 rods). No changes to the assumed number of failed fuel rods or assumed radial peaking factor are required to support the MUR power uprate. As part of the cycle reload safety evaluation process, the continued applicability of the gap fractions for non-LOCA events is verified per Regulatory Guide 1.183, Table 3, footnote 11. The release pathways, X/Qs, and dose conversion factors are unchanged from the AST license amendment requests and associated SERs (References II-23 and II-24). Therefore, the current fuel handling accident dose evaluation remains bounding for the MUR power uprate.

II.2.28 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) - UFSAR 15.4.6

UFSAR Section 15.4.6 describes the rupture of a CRDM housing (RCCA Ejection) accident, which was analyzed for RCS overpressure, peak clad temperature, percent fuel melt, and average fuel enthalpy. The analysis was performed to support the transition to AREVA AMBW fuel. This analysis was consistent with the NRC approved rod ejection topical report (Reference II-20) assumptions, methodology, and calculational techniques. The rod ejection analysis was described in the license amendment request to use AREVA AMBW fuel (Reference II-17) and was NRC approved in References II-5 and II-6.

Rod ejection analyses are performed at hot zero power and hot full power. A point kinetics RETRAN analysis is performed at nominal hot full power conditions and a hot spot RETRAN analysis is performed at deterministic hot full power conditions (102% of 2893 MWt). The hot spot analysis model used a nominal core power of 2893 MWt with an FQ of 2.51, which included the 2% power calorimetric uncertainty, to determine the initial hot spot model power level. Thus, the hot spot analysis accounts for 2% power uncertainty above 2893 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

UFSAR Section 15.4.6.2.1.3 describes a RCS overpressure analysis. Section 2.2.5 of VEP-NFE-2-A (Reference II-20) refers to a conservative, generic overpressure analysis performed by Westinghouse and documented in WCAP-7588, Revision 1-A (Reference II-8). WCAP-7588, Revision 1-A, Sections 2.6 and 4.4 describe the methodology and conservative analysis. The generic analysis reactivity assumptions remain bounding for 2940 MWt. Therefore, the rod ejection accident is bounding for the MUR power uprate.

II.2.29 Radiological Consequences

II.2.29.1 LOCA Dose Evaluation - UFSAR 15.4.1.8

As discussed in UFSAR Section 15.4.1.8, the LOCA event analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The existing NAPS Technical Specification 4.3.1.2 restricts fuel enrichment to 4.6 w/o U-235, which is unchanged by the MUR power uprate. Fuel assembly exposure is restricted to a lead rod burnup of 62,000 MWD/MTU. The MUR power uprate results in limited changes to core power, burnup history, and enrichment. Thus, the source term core inventory incorporated into the existing LOCA dose analysis remains bounding.

The current LOCA dose analysis is based on a core inventory that assumes 2958 MWt, which is 102.2% of 2893 MWt. The LOCA radiological consequences result from the release of the core inventory to the RCS and then to the environment. The release pathways, X/Qs, and dose conversion factors are unchanged from the AST and Generic Safety Issue 191 license amendment requests and associated SERs (References II-23 through II-26). Therefore, the existing LOCA radiological analysis remains bounding for the MUR power uprate.

II.2.29.2 Locked RCP Rotor Dose Evaluation - UFSAR 15.4.4.3

As discussed in UFSAR Section 15.4.4.3, the locked RCP rotor event analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current locked RCP rotor analysis is a function of core power, enrichment, burnup, gap fractions for non-LOCA events from Regulatory Guide 1.183, an assumed percent of failed fuel, and an assumed radial peaking factor. The existing locked RCP rotor dose evaluation was performed using the core inventory that assumes 2958 MWt, which is 102.2% of 2893 MWt. No changes to the assumed percent of failed fuel or assumed radial peaking factor are required to support the MUR power uprate. The steam release modeled in the current locked RCP rotor analysis is consistent with a core thermal power of 2951 MWt (102% of 2893 MWt). The release pathways, X/Qs, and dose conversion factors are unchanged from the AST license amendment requests and associated SERs (References II-23 and II-24). Therefore, the current locked RCP rotor dose evaluation remains bounding for the MUR power uprate.

II.2.29.3 Fuel Handling Accident Dose Evaluation

The fuel handling accident dose evaluation is discussed in Section II.2.27.

II.2.29.4 Main Steam Line Break Dose Evaluation

The current MSLB radiological analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The analysis involves primary coolant radiological source release to the secondary side from the SG and then to the environment. The source terms from UFSAR Section 11.1.1.2 for equilibrium conditions with 1% failed fuel are normalized to the Technical Specifications Dose Equivalent Iodine-131 limits in the primary coolant, which essentially removes the power dependence from the analysis.

The steam releases modeled in the current MSLB analysis are consistent with a core thermal power of 2951 MWt (102% of 2893 MWt). The release pathways, X/Qs, and dose conversion factors are unchanged from the AST license amendment request and associated SER (References II-23 and II-24). Since the Technical Specification Dose Equivalent Iodine-131 limits, steaming rates, and other key dose parameters are not changing as a result of the MUR power uprate, the existing MSLB radiological analysis remains bounding for the MUR power uprate.

II.2.29.5 Steam Generator Tube Rupture Dose Evaluation

The current steam generator tube rupture (SGTR) radiological analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The analysis involves primary coolant radiological source release to the secondary side from the SG and then to the environment. The source terms from UFSAR Section 11.1.1.2 for equilibrium conditions with 1% failed fuel are normalized to the Technical Specifications Dose Equivalent Iodine 131 limits in the primary coolant, which essentially removes the power dependence from the analysis.

The steam releases modeled in the current SGTR analysis are consistent with a core thermal power of 2951 MWt (102% of 2893 MWt). The release pathways, X/Qs, and dose conversion factors are unchanged from the AST license amendment request and associated SER (References II-23 and II-24). Since the Technical Specification Dose Equivalent Iodine 131 limits, steaming rates, and other key dose parameters are not changing as a result of the MUR power uprate, the existing SGTR radiological analysis remains bounding for the MUR power uprate.

II.2.29.6 Volume Control Tank Rupture Dose Evaluation

The volume control tank rupture dose evaluation is discussed in Section II.2.20.

II.2.29.7 Waste Gas Decay Tank Rupture Dose Evaluation

The waste gas decay tank rupture dose evaluation is discussed in Section II.2.19.

II.2.30 Natural Circulation - UFSAR 15.2.9

Natural circulation is analyzed in two events: loss of offsite power and Appendix R safe shutdown.

The loss of offsite power analysis used the NRC approved RETRAN analysis methodology (Reference II-18) to determine the plant transient. The analysis of record showed that natural circulation flow was sufficient to provide adequate core decay heat removal to prevent fuel or clad damage following a reactor trip and RCP coastdown. The analysis assumed an initial core power of 2951 MWt, or 102% of 2893 MWt.

The NAPS Appendix R Report, Section 3.5.1 states that one RCS loop is required to ensure that natural circulation can be established and maintained. A review of safety analysis calculations referenced in the NAPS Appendix R Report, confirmed that the safe shutdown systems provide adequate natural circulation cooling after the MUR power uprate.

II.2.31 LOCA Mass and Energy Release - UFSAR 6.2.1

II.2.31.1 Long-term LOCA Mass and Energy Release Analysis

The long-term LOCA mass and energy releases used in the UFSAR Chapter 6 containment analyses were submitted to the NRC in Reference II-16. The NRC approved these LOCA containment analyses in Reference II-9. Westinghouse mass and energy release analyses for the blowdown and reflood phases used NRC approved methods and assumed a core power of 102.6% of 2893 MWt. The GOTHIC post-reflood mass and energy releases were generated with NRC approved methods, assuming 102% of 2893 MWt. LOCA mass and energy releases remain bounding for the MUR power uprate conditions. The UFSAR LOCA containment response analyses remain bounding. The analyses confirmed that, after a LOCA, the net positive suction head available for the recirculation spray and low head safety injection pumps during sump recirculation was not affected by the MUR power uprate.

II.2.31.2 Short-term LOCA Mass and Energy Release Analysis

UFSAR Sections 6.2.1.1.2 and 6.2.1.3.2 describe the analyses of containment subcompartment response post-LOCA. NAPS has been approved for leak before break methods. The only break locations that need consideration are the largest primary loop piping branch lines off of the primary loop piping. The short-term LOCA mass and energy releases in UFSAR Tables 6.2-6 (150 in^2 cold leg limited displacement rupture), 6.2-7 (surge line double-ended rupture), 6.2-8 (spray line double-ended rupture) and 6.2-9 (hot leg single-ended split) have not changed since original NAPS licensing. The short-term LOCA mass and energy releases are affected by changes in RCS temperatures, due to the fluid density effect on the initial pressure pulse created when the pipe ruptures. The power uprate design RCS temperatures were reviewed and confirmed to be bounded by the inputs to the existing short term mass and energy release analysis. The short-term LOCA mass and energy releases were generated at 102.2% of 2893 MWt. Therefore, the UFSAR containment short-term subcompartment analyses remain bounding for the MUR power uprate.

II.2.32 Main Steam Line Break Mass and Energy Release

The MSLB mass and energy releases used in the UFSAR Chapter 6 containment analyses were described to the NRC in Section 3.2.2 of Reference II-16. The NRC approved the MSLB containment analyses in Reference II-9. Westinghouse mass and energy release analyses used NRC approved methods, assuming 102.2% of 2893 MWt. The containment results in UFSAR Table 6.2-16 (Summary of Main Steam Line Break Peak Pressure and Temperature) are conservative and bounding for the MUR power uprate. The UFSAR conclusions remain valid for the long-term steam line break event inside containment.

II.2.33 ATWS/AMSAC - UFSAR 4.3.1.7 and 7.7.1.14

Anticipated transient without scram (ATWS) mitigation system activation circuitry (AMSAC) has been incorporated into the NAPS Units 1 and 2 plant designs per 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transient Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plant." The NAPS AMSAC system automatically initiates a turbine trip and starts Auxiliary Feedwater (AFW) pumps under conditions indicative of an ATWS and a loss of main feedwater. AMSAC was described in an NRC submittal (Reference II-22). Section F, "Operating Bypass," of Reference II-22 confirmed that the Westinghouse generic analyses in Reference II-21 were applicable to NAPS. The NRC approved the NAPS AMSAC design in Reference II-12.

The AMSAC system and the analyses in Reference II-21 were reviewed with respect to the proposed MUR power uprate. NAPS Units 1 and 2 are 3-loop PWRs with model 54F replacement SGs, which are similar to the 3-loop plant model in the generic analyses. The key differences are core power and total primary system relief valve capacity (pressurizer safety valves and power operated relief valves). This can be expressed as a ratio of core power to relief capacity in MWt/lbm/hr. The NAPS ratio of core thermal power (2951 MWt or 102% of 2893 MWt) to relief valve capacity is less than the MWt/lbm/hr ratio used in the 3-loop case in Reference II-21. Therefore, the generic 3-loop PWR analyses in Reference II-21 remain bounding for NAPS at 102% of 2893 MWt and also bound the MUR power level. The generic 3-loop PWR analyses indicated more than 400 psi margin to the ASME Boiler and Pressure Vessel (B&PV) Code Service Level C acceptance criterion of 3200 psig.

The NAPS AMSAC design specifies a nominal permissive C-20 setpoint based on the generic 40% setpoint minus an allowance for inaccuracies in the turbine first stage impulse pressure channels. The turbine first stage impulse pressure channels will be re-scaled due to a higher MUR power uprate full power steam flow. There are no other AMSAC impacts as a result of the proposed power uprate.

II.2.34 Station Blackout - UFSAR 8.1.2

Station blackout (SBO) is discussed in Section V.1.B.

II.2.35 Analyses to Determine EQ Parameters - UFSAR 3.11

Critical Environmental Qualification (EQ) parameters include temperature, pressure, radiation, relative humidity, pH and submergence. Current analyses for long-term LOCA and steamline break mass and energy releases remain bounding at the power uprate conditions. The temperature, pressure, relative humidity, pH and submergence conditions are therefore bounding for the proposed uprate. Radiological doses used in the EQ evaluations do not bound the increase in doses due to the power uprate. Therefore, Dominion concludes that, with the exception of radiological doses, the current EQ parameters remain bounding for the MUR power uprate. The evaluation for radiological effects is discussed in Section III.2.

EQ of electrical equipment is discussed in Section V.1.C.

II.2.36 Safe Shutdown Fire Analysis (Appendix R report) - UFSAR 9.5.1

UFSAR Section 9.5.1 describes the fire protection system and design bases for compliance with 10 CFR 50, Appendix R. The NAPS Appendix R Report describes the system functions that ensure safe shutdown is achieved after a fire. NAPS Appendix R Report, Section 3.9 identifies the calculations that provide the technical basis for the Appendix R fire protection program. Reviews concluded that the calculations cited in the Appendix R Report remain bounding for the MUR power uprate. Additional calculations that are not cited in the Appendix R Report, but have provided a basis for the program were reviewed and remain bounding for the MUR power uprate.

The safe shutdown analyses that support the Appendix R program were reviewed. The analyses support a core power of 2951 MWt, or 102% of the current RTP of 2893 MWt. The power uprate does not change the design, function or impose any new requirements on the systems or components that support the Appendix R safe shutdown requirements (e.g., residual heat removal, chemical and volume control). Operator actions in response to an Appendix R fire are not adversely impacted. The MUR power uprate does not affect the worst case fire location or the post-fire local operations and capability to complete repairs. The worst case fire scenario timeline indicates that the plant can achieve cold shutdown within the 72-hour requirement. The 72-hour cooldown requirement in 10 CFR 50, Appendix R, Sections III.G.1.b and III.L is met. Therefore, the Appendix R safe shutdown analyses remain bounding for the MUR power uprate.

II.2.37 Spent Fuel Pool Cooling - UFSAR 9.1.3

The NAPS UFSAR outlines the cooling requirements for the SFP. Three scenarios are described. Each scenario assumes that fuel movement begins no earlier than 100 hours after the reactor is subcritical. First – normal back-to-back scenario, where the most recent previous refueling occurred within the last 120 days prior to a full core offload. Second – normal non back-to-back scenario, where the most recent previous refueling occurred more than 120 days prior to a full core offload. Third – abnormal back-to-back scenario, where an unscheduled shutdown of one unit, which requires a full core offload, occurs after the other unit has returned to operation following back-to-back refuelings. The design basis SFP temperature limit for the first and second scenarios is 140°F; the third scenario limit is 170°F. The SFP heat loads in the analyses of record were calculated assuming 2% calorimetric uncertainty. The SFP cooling system is capable of maintaining the SFP temperature below the design basis limits with the above heat loads. Therefore, there is no change to the loss of cooling analysis.

Several secondary assumptions in the heat load analysis of record were also evaluated. A rated thermal power increase will increase average assembly burnup. Second and third cycle fuel burnups were originally assumed in the SFP heat load analysis of record. These values are compared, on a reload basis, to batch averaged burnups for twice and thrice burned fuel being discharged. Based on these burnups, at least 3% margin exists to the assumed burnups. It is possible that the MUR power uprate may require an additional four fresh assemblies (68 total) to meet optimum cycle energy requirements. The spent fuel pool heat load analysis supports fresh batch sizes of 64 ± 4 assemblies. The power uprate does not impact other secondary assumptions such as assembly loading (in MTU), assembly enrichment, outage duration, offload time, and cycle length (in EFPD).

In conclusion, the spent fuel pool heat load assumptions included in the NAPS UFSAR will remain bounding for the MUR power uprate. Refer to Section VI.1.D for further discussion on the spent fuel pool storage and cooling.

II.2.38 Internal Flooding - UFSAR 2.4.10

The design bases for flooding inside and outside the containment building were evaluated. The power uprate results in increased piping system flowrates (e.g., condensate, main feedwater and main steam). These changes were evaluated to determine any impact on the flooding analysis. Based on flooding analysis calculation reviews, it was determined that the current flood levels are not affected by the MUR power uprate.

II.3 Design Transients

II.3.1 Nuclear Steam Supply System Design Transients

NSSS design transients were specified in the original design analyses of NSSS components cyclic behavior. The selected transients are conservative representations of transients that when used as a basis for component fatigue analysis, provide confidence that the component is appropriate for its application over the 60-year plant license period. The RCS and its auxiliary system components are designed to withstand the cyclic load effects from RCS temperature and pressure changes. The existing design transients were evaluated for their continued applicability at MUR power uprate conditions.

The key plant design parameters for the NSSS design transients are RCS hot and cold leg temperatures (T_{hot}, T_{cold}), secondary side steam temperature and pressure (T_{steam}, P_{steam}), and the secondary side feedwater temperature. The existing design transients for parameters except feedwater temperature bound plant operation at the uprated conditions. Those design transients with feedwater temperature variation required revision. This change was the result of the uprated full power feedwater temperature increase of 9°F. Note that the previous design transient analysis was generic, and was revised to represent a NAPS specific analysis. Some conservatism was removed from the existing design transients so they would better represent uprated plant conditions. These revised design transients were considered in the various NSSS component evaluations to ensure the component fatigue analyses were satisfactory. The component fatigue evaluation results are discussed in Section IV.

The primary to secondary differential pressure limit was not exceeded for any normal or upset design transient. The frequencies of occurrence for the 60-year plant licensed period are unchanged for the power uprate. No new design transients are created as a result of the MUR power uprate.

II.3.2 Auxiliary Equipment Design Transients

The NAPS auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the RCS, and certain NSSS auxiliary systems piping, heat exchangers, pumps and tanks. The transients are sufficiently conservative, such that when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the plant operating license period.

The only auxiliary equipment design transients potentially impacted by the power uprate are those transients associated with full load NSSS design temperatures (T_{hot} and T_{cold}). These temperature transients are defined by the differences between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. Since the operating coolant temperatures in the auxiliary systems are not impacted by the power uprate, the temperature difference between auxiliary systems and the RCS loops is only affected by changes in the RCS operating temperatures. The transients assume a full load NSSS T_{hot} and T_{cold} of 630°F and 560°F, respectively. These full load temperatures were selected for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters. The approved NSSS design temperature range for T_{hot} and T_{cold} used to develop the current design transients is smaller than the reference design values. The smaller full load temperatures from the MUR power uprate result in less severe design temperature transients. Therefore, the existing auxiliary equipment design transients are conservative and bounding for the power uprate.

II.3.3 Plant Operability

The pressure control component sizing and plant operability for normal condition transients were evaluated for NAPS.

RCS pressure control component sizing includes the pressurizer heater, spray, and PORV capacities. These components must continue to successfully perform their intended functions. Plant operability for Condition I (normal condition) transients includes the plant response to 5-percent/minute loading and unloading, 10-percent step-load increase or decrease, and large-load rejection. These transients must not result in a reactor trip, engineered safety features actuation system (ESFAS) or challenge the pressurizer or main steam safety valves. This analysis was conducted to confirm the continued plant acceptability to meet these requirements at power uprate conditions.

Pressure control component sizing and plant operability for normal condition transients were reviewed independently. The reviews concluded that the power uprating does not result in unacceptable plant operations. The existing pressure control components (heater, spray, and PORV) meet the sizing criteria at the uprated conditions. The component capacities are adequate to mitigate the sizing basis transients without exceeding the limits. Adequate margin exists to relevant reactor trip and ESFAS setpoints during the normal condition transients at uprated power conditions. The control systems remain stable and support the power uprate for normal condition transients. The existing setpoints for the reactor control, pressurizer pressure control, pressurizer level control, steam generator level control, and steam dump control remain valid.

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II-27 Letter from Robert A. Clark (USNRC) to W.L. Stewart (Virginia Power), *Fire Protection Rule – Alternate Safe Shutdown Capability – Sections III.G.3 and III.L of Appendix R to 10 CFR 50 – North Anna Power Station, Units Nos. 1 and 2 (NA-1&2)*, November 18, 1982.

II-28 Letter from Thomas M. Novak (USNRC) to W.L. Stewart (Virginia Power), *Technical Exemptions from Appendix R, 10 CFR Part 50/North Anna Power Station, Units Nos. 1 and 2 (NA-1&2)*, November 6, 1986.

II-29 Letter from Robert A. Clark (USNRC) to W.L. Stewart (Virginia Power), *Justification For Continued Operation of the North Anna Power Station, Units Nos. 1 & 2 (NA-1&2) Environmental Qualification of Safety Related Equipment (Category II.B)*, March 23, 1983.

II-30 Letter from James R. Miller (USNRC) to W.L. Stewart (Virginia Power), *Environmental Qualification of Electric Equipment (10 CFR 50.49) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2)*, December 11, 1984.

II-31 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*, ML030160804, ML030160825 and ML030160848, December 2002.

III. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling and flooding).

2. For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:

   A. Identify the transients/accidents that is the subject of the analysis

   B. Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate

   C. Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59.

   D. Provide a reference to the NRC's approval of the plant's reload methodology

3. For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should include:

   A. Identify the transient or accident that is the subject of the analysis

   B. Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate

   C. Confirm that the limiting event determination is still valid for the transient or accident being analyzed

   D. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies.

E. Provide references to staff approvals of the methodologies in Item D. above

F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology

G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate

H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head, valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from previous analysis

RESPONSE TO III - ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

III.1 UFSAR Section 15.4.1 Loss of Reactor Coolant From Ruptured Pipes or From Cracks in Large Pipes Including Double Ended Rupture That Actuates the Emergency Core Cooling System (Large Break Loss of Coolant Accident)

NAPS UFSAR Sections 15.4.1.10 through 15.4.1.17 describe the large break LOCA analysis for AREVA AMBW fuel. The analyses use the realistic large break LOCA analysis methodology (Reference III-1) for calculation of fuel PCT and oxidation (local and whole-core). The realistic large break LOCA analyses were NRC approved with the AMBW fuel product implementation (Reference III-2 for Unit 1 and Reference III-3 for Unit 2). Since NRC approval, the realistic large break LOCA analyses have been augmented by evaluations under 10 CFR 50.46. Items III.3.C, III.3.G and III.3.H above are unchanged, because this was not a reanalysis.

The realistic large break LOCA analysis uses a single power level with an uncertainty range that is sampled per the Reference III-1 methodology. The analysis of record used a single core power level of 2893 MWt with a sampled normal uncertainty distribution with a one sigma (1σ) standard deviation of 1.1%. The Large Break LOCA was determined to be an accident for which the existing analysis of record does not bound plant operation at the proposed uprated power level. Specifically, the proposed RTP of 2940 MWt exceeds the nominal core power level of 2893 MWt that was assumed in the realistic large break LOCA analysis of record.

MUR Evaluation Methodology

For the MUR power uprate, a sensitivity study of the realistic large break LOCA analysis of record was performed to assess the change in PCT and oxidation due to the increase in core rated thermal power and decrease in power calorimetric uncertainty. The realistic large break LOCA analysis set of 59 cases was repeated with only two changes related to core power:

- The nominal core power was increased from 2893 MWt to 2942.2 MWt.
- The one sigma (1σ) standard deviation was reduced from 1.1% to 0.2% based on the new calorimetric uncertainty calculation using ultrasonic feedwater flow meters.

The core power of 2942.2 MWt was selected as the upper bound rated thermal power assuming a 0.3% power calorimetric uncertainty with ultrasonic feedwater flow meters. The power level of 2942.2 MWt was selected to be conservative compared to the licensed core power level of 2940 MWt that was determined later by the revised power calorimetric uncertainty analysis. The one sigma (1σ) standard deviation of 0.2% was selected to bound the two sigma (2σ) 0.3% calorimetric uncertainty. The single point power level analysis with a ranged uncertainty is consistent with the NAPS realistic large break LOCA analysis licensing basis. Table III-1 compares the analysis power input parameters.

Table III-1
Comparison of Realistic Large Break LOCA Inputs Related to Core Power

Parameter	Analysis of Record	MUR Sensitivity Studies
Core Power, MWt	2893	2942.2
Power Operational Range, MWt	±0.0	±0.0
1σ Standard Deviation, %/MWt	1.1/31.8	0.2/5.9
Probability Distribution	Normal	Normal

The realistic large break LOCA sensitivity study was based on the analysis of record that applied the current licensed methodology in Reference III-1. Separate calculations were performed for Units 1 and 2 consistent with the UFSAR. The core power uncertainty was sampled using a nominal value of 2942.2 MWt and one sigma (1σ) standard deviation of 0.2%. No other input parameters were changed. This method of changing only the plant parameter of interest is consistent with previous evaluations for North Anna. For example, an evaluation of the sensitivity of refueling water storage tank temperature was performed with this method and submitted to the NRC in Reference III-5.

By running the 59 cases with the same inputs (and no change to the seed) as the analysis of record except for core power, the effect of the MUR power uprate and reduced power calorimetric uncertainty could be quantified. Also, the case results would provide a large sample to assess whether PCT and core power are correlated over the sampling range for power. Previous AREVA realistic large break LOCA studies had identified that power is not a sufficiently influential parameter that dominates over other sampled parameters.

Evaluation of Sensitivity Study Results

The following observations were made from the 59 cases that were run for each North Anna unit:

- There was no observed correlation between the changes in PCT and the changes in core power inputs evaluated.
- The sensitivity studies confirmed that the maximum PCT cases that are identified in the UFSAR (case 28 for Unit 1 and case 8 for Unit 2) have not changed.
- The change in PCT is small for the change in core power parameters. For the analysis of record maximum PCT cases identified in the UFSAR, the Unit 1 PCT increased by 2°F and the Unit 2 PCT increased by 20°F. The PCT changes are less than the 30°F that is specified as the threshold for low importance ranking in Appendix C of EMF-2103(P)(A) (Reference III-1).

Evaluation of 10 CFR 50.46 Acceptance Criteria

Tables III-2 and III-3 summarize the change in results for the limiting realistic large break LOCA analyses of record for NAPS Units 1 and 2, respectively. The tables report the change in results for the maximum PCT cases that are reported in the NAPS UFSAR. Reporting the change in PCT for the maximum PCT case has been used previously to document the effect of plant model changes. The following conclusions were made with respect to 10 CFR 50.46 acceptance criteria:

- The change in PCT is small for the current analysis of record maximum PCT case reported in the UFSAR. Unit 1 PCT increased by 2°F, and Unit 2 PCT increased by 20°F. Considering the existing PCT results, including the assessments against the analysis of record in UFSAR Table 15.4-27, significant margin remains to the 2200°F limit as shown in Table III-4.

- The maximum increase in local clad oxidation was 0.19%, as shown in Tables III-2 and III-3. Considering the existing local oxidation results, including the assessments against the UFSAR analysis of record, margin remains to the 17% limit.
- The change in core-wide oxidation is small. Considering the existing core-wide oxidation results, including the assessments against the UFSAR analysis of record, margin remains to the 1% limit.

Therefore, the requirements of 10 CFR 50.46 continue to be met for the MUR power uprate. While the changes in PCT are small, a penalty to reported PCT will be applied to the analysis of record to address the potential for cumulative effects of small changes. The observed penalties due to the MUR power uprate from the limiting realistic large break LOCA cases will be incorporated into the analysis basis with the power uprate implementation.

Table III-2
Change to North Anna Unit 1 Realistic Large Break LOCA Analysis Results

Parameter	Analysis of Record (AOR)	MUR Sensitivity Study	MUR – AOR
PCT, °F	1853	1855	+ 2
Local Oxidation, %	2.61	2.75	+ 0.14
Whole-Core Oxidation, %	0.03	0.03	+ 0.00

Table III-3
Change to North Anna Unit 2 Realistic Large Break LOCA Analysis Results

Parameter	Analysis of Record (AOR)	MUR Sensitivity Study	MUR – AOR
PCT, °F	1789	1809	+ 20
Local Oxidation, %	1.75	1.94	+ 0.19
Whole-Core Oxidation, %	0.04	0.05	+ 0.01

Table III-4
Summary of Large Break LOCA PCT Results with MUR Penalty

Parameter	Current LBLOCA PCT from UFSAR Table 15.4-27	MUR Penalty	Revised LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements
Unit 1 PCT, °F	1925	+2	1927
Unit 2 PCT, °F	1919	+20	1939

III.2 Analysis to Determine EQ Parameters Radiological Effects - UFSAR 3.11

III.2.A Normal Operation

Normal non-radiological plant operating conditions assumed within all environmental zones (i.e., temperature, pressure, humidity) remain unchanged for power uprate operation. A separate evaluation was performed to assess potential increase in normal operation radiation dose used in the EQ program. In general, power uprate operation would be expected to increase the core inventory of radioisotopes by the percentage increase in core power and potentially to increase the normal operation radiation source term. However, this potential increase in radiation source term will not affect the currently estimated normal operation doses used for EQ, because of several conservative factors incorporated into the current estimates. The most significant of these considerations are: a) use of a dose for a given radiation zone designation that represents the maximum end of the normal operation range and b) the limitation imposed by plant operations as a result of Technical Specifications limits on RCS coolant activity (i.e., allowable limits of operation are approximately one-third of the value associated with the assumed 1% fuel defects used in the normal operation source term). The conditions used in the EQ program for normal operation therefore remain bounding for the MUR power uprate, with the exception of the dose levels for the reactor vessel excore neutron detectors. The excore detectors radiation dose increases such that the EQ in-service life may be decreased. These excore detectors were recently replaced on both units. Prior to operating above the 2893 MWt (98.4% RTP), Dominion will determine the EQ service life of the excore detectors. A calculation is being developed to evaluate the dose impact on these detectors. Preliminary results indicate no impact on radiation dose margin or qualified life.

III.2.B Accident Conditions

There is no change in assumed accident temperature, pressure, or humidity due to power uprate operation. The post accident (i.e., LOCA) radiation effects have been updated to reflect the power uprate conditions. The evaluation details are provided below.

The current post accident dose estimates utilized for EQ are based on LOCA and radiation source terms corresponding to a core power level of 2900 MWt, assuming a 12 month fuel cycle and the ACTIVITY2 computer code. These were the design basis calculations from original plant licensing. For the MUR power uprate, the applicable assumptions of the post-accident radiation source terms are a core power of 2951 MWt and 18 month fuel cycle. The computer code used to develop the core inventory applicable for the MUR uprate is ORIGENS.

In Section 2.3.1 of the Millstone 3 Stretch Power Uprate licensing amendment request (Reference III-4) scaling factors were developed that accounted for an increase in core power (3636 to 3723 MWt), an 18 month versus 12 month fuel cycle, and current use of the ORIGENS computer code versus the ACTIVITY2 code used in the original design basis analysis. For the NAPS MUR power uprate, an evaluation was performed to confirm applicability of the scaling factors developed for Millstone 3 is conservative for NAPS. The resulting factors were used to modify the existing post-accident total integrated radiation dose for all environmental zones identified in the NAPS EQ program. These augmented values represent the MUR power uprate radiation environment considered for EQ. Table III-5 provides a summary of the current and revised radiation parameters, for each separate class of equipment that is monitored within the EQ program.

Table III-5
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
01.1	5	ITE IMPERIAL CORP.	K-LINE BREAKER	AB-280	1.32E+03	5.90E+04	6.03E+04	7.08E+04	7.21E+04	1.00E+05	28%
02.5	0	MJ ELECTRIC, INC.	TYPE 2	AB-280	1.32E+03	2.85E+04	2.98E+04	3.42E+04	3.55E+04	1.00E+05	> 50%
03.1	25	RELIANCE ELECTRIC CO./LIMITORQUE	SMB00	AB-2C	4.20E+06	8.00E+06	1.22E+07	9.60E+06	1.38E+07	2.00E+07	31%
03.3	9	RELIANCE ELECTRIC CO./LIMITORQUE	P, .70 HP,RH-INSUL, 10FT#, SMB00	RC-241A, 262A, 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.04E+08	> 50%
04.1	9	WESTINGHOUSE ELECT CORPORATION	HSDP	AB-2C	4.20E+06	8.00E+06	1.22E+07	9.60E+06	1.38E+07	5.12E+07	> 50%
04.2	12	GENERAL ELECTRIC CO.	5K6328XC264A/ 5K509DT6488M	SFGD-1	3.90E+02	7.00E+06	7.00E+06	8.40E+06	8.40E+06	5.50E+07	> 50%
04.3	10	WESTINGHOUSE ELECT CORPORATION	ABDP	SFGD-1	3.90E+02	7.00E+06	7.00E+06	8.40E+06	8.40E+06	1.40E+07	40%
04.4	10	GENERAL ELECTRIC CO.	5K6319XJ1D	SFGD-1	5.25E+04	7.50E+06	7.55E+06	9.00E+06	9.05E+06	2.00E+08	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
04.7	5	DELPHI CONTROL SYSTEMS, INC	11705	SFGD-1 QSPA-271B	3.90E+02	7.00E+06	7.00E+06	8.40E+06	8.40E+06	1.00E+07	16%
04.9	4	FRANKLIN ELECTRIC CO., INC. (MOTORS)	131300711	QSPA-271B	3.90E+02	5.40E+06	5.40E+06	6.48E+06	6.48E+06	8.70E+06	26%
06.01	5	BIW CABLE SYSTEMS(BOSTON INSULATED WIRE)	NGA, NGB, NGC	RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.89E+07 Note 1	1.00E+08	11%
06.02	5	ROCKBESTOS CO.	NGA, NGB	RC-262A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.89E+07 Note 1	1.96E+08	> 50%
06.03	6	ROCKBESTOS CO.	NGA-15	RC-262A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.89E+07 Note 1	2.00E+08	> 50%
06.05	5	GENERAL CABLE CO.	NGB	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.89E+07 Note 1	2.00E+08	> 50%
06.06	5	GENERAL CABLE CO.	NGA, NGB	SFGD-1	3.90E+02	7.00E+06	7.00E+06	8.40E+06	8.40E+06	2.00E+08	> 50%
06.07	6	OKONITE COMPANY	NGA-20	RC-262A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.88E+07 Note 1	2.00E+08	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
06.08	5	OKONITE COMPANY	NGA-01-01	RC-262A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.88E+07 Note 1	2.00E+08	> 50%
06.09	7	OKONITE COMPANY	NGA-10	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.74E+07 Note 1	2.00E+08	> 50%
06.10	2	CONTINENTAL WIRE & CABLE	NGB-63, NGB-65	RC-291A RC-262A RC-241A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%
06.11	5	RAYCHEM CORP	NGC-10, NJK-64,	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	7.51E+07 Note 1	2.00E+08	> 50%
06.13	5	BRAND-REX COMPANY	NGA-34,	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	9.14E+07 Note 1	2.00E+08	> 50%
06.16	6	VALIDYNE ENGINEERING CORP	MC370AD-series & other components	RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.20E+08	> 50%
06.16	6	VALIDYNE ENGINEERING CORP	MC370AD-series & other components	AB-259B	1.32E+03	3.10E+04	3.23E+04	3.72E+04	3.85E+04	2.00E+05	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
06.17	1	GENERAL ELECTRIC CO.	NGE-SIS	RC-216A	4.50E+05	1.80E+07	1.85E+07	2.16E+07	2.21E+07	4.00E+07	45%
06.19	1	ROCKBESTOS CO.	EGS Component	AB-291C RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%
06.20	0	OKONITE COMPANY	5 kV Cable - black EPR	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.74E+07 Note 1	2.00E+08	> 50%
08.01	8	CONAX BUFFALO CORP.	7C47-10000-01	RC-262A, RC-241A, RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.20E+08	> 50%
08.03	9	GEMS SENSORS DIV/IMO INDUSTRIES	XM-54853-56-1000	RC-216A/B, Surry RC-27B	5.25E+04	3.80E+07	3.81E+07	4.56E+07	4.57E+07	2.00E+08	> 50%
08.05	33	ROSEMOUNT, INC. (Inside containment)	01153	RC-241B, 262B, 291B	5.25E+04	6.79E+06	6.84E+06	8.15E+06	8.20E+06	5.00E+07	> 50%
08.07	19	ROSEMOUNT, INC. (outside containment)	01153	Surry's AB-02C	4.20E+06	8.00E+06	1.22E+07	9.60E+06	1.38E+07	2.21E+07	38%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
08.08	7	GAMMA-METRICS Note: These components are currently being replaced to achieve plant life extension. Completed evaluation indicates no issues with margin.	200112-008 and other components	SUM-1							See note at left
08.09	9	WESTINGHOUSE ELECT CORPORATION	2654C65G01	RC-241A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.66E+07 Note 1	1.60E+08	46%
08.10	9	VICTOREEN	877-1/878-1-5	RC-291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.20E+08	> 50%
08.22	7	ITT Barton	752	AB-259B	7.95E+03	3.10E+04	3.90E+04	3.72E+04	4.52E+04	1.00E+05	> 50%
08.24	24	WEED INSTRUMENT CO., INC.	N9001D-2B	SUM-1	1.95E+08	6.80E+02	1.95E+08	8.16E+02	1.95E+08	3.03E+08	36%
08.25	8	ENDEVCO (DIV OF ALLIED SIGNAL INC)	2273AM1	SUM-1	4.50E+07	1.80E+07	6.30E+07	2.16E+07	8.35E+07	2.22E+08	> 50%
08.26	8	ROSEMOUNT, INC.	01154- / 1153	RC-262B	5.25E+04	6.79E+06	6.84E+06	8.15E+06	8.20E+06	5.55E+07	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
09.1	26	NAMCO CONTROLS (DIV OF ACME-CLEVELAND)	EA180-31302	RC-241A, 262A, 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.04E+08	> 50%
11.3	6	RELIANCE ELECTRIC CO.	21-1A-450	SFGD-1	3.90E+02	7.00E+06	7.00E+06	8.40E+06	8.40E+06	2.04E+08	> 50%
15.1	12	CONAX BUFFALO CORP.	7057-10000 series	RC-262B, AB-259B	5.25E+04	7.40E+06	7.45E+06	8.88E+06	8.93E+06	1.00E+08	> 50%
15.4	0	CONAX BUFFALO CORP.	7V26-10000 series	RC-241B	4.50E+04	6.79E+06	6.84E+06	8.15E+06	8.19E+06	1.90E+08	> 50%
16.2	11	RAYCHEM CORP	WCSF/NJRT	RC-241A, 262A, 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	1.51E+08 Note 1	2.15E+08	30%
16.4	5	OKONITE COMPANY	T-35/T95	RC-241A, 262A, 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	1.78E+08 Note 1	2.00E+08	11%
16.5	6	RAYCHEM CORP		AB-2C	4.20E+06	8.00E+06	1.22E+07	9.60E+06	1.38E+07	5.00E+07	> 50%
17.1	8	CONNECTRON INC	NSS3	SUM-1	5.25E+04	6.79E+06	6.84E+06	8.15E+06	8.20E+06	2.50E+07	> 50%
17.2	8	GENERAL ELECTRIC CO.	EB-5/EB-25/EB-27	SUM-1	1.95E+08	6.80E+02	1.95E+08	8.16E+02	1.95E+08	2.20E+08	11%
17.3	8	MARATHON ELEC. MFG. CORP.	200/1500 SERIES	RC-291B	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.04E+08	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
34.1	8	CONAX BUFFALO CORP.	N-11000 SERIES	SUM-1	1.95E+08	6.80E+02	1.95E+08	8.16E+02	1.95E+08	2.25E+08	13%
34.2	7	CONAX BUFFALO CORP.	PL SERIES/4P	SUM-1	1.95E+08	6.80E+02	1.95E+08	8.16E+02	1.95E+08	2.25E+08	13%
34.3	5	ROSEMOUNT, INC.	353C	RC-241A, 262A, 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	1.10E+08	39%
34.4	6	EGS, A DIVISION OF SAIC	EGS/PATEL QDC	RC-241A, 262A and 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%
34.5	6	EGS, A DIVISION OF SAIC	GB-1/GB-2/GB-3	RC-241A, 262A and 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%
35.1	31	AUTOMATIC SWITCH CO/ASCO	NPKX8321A1V1 0688	RC-241A, 262A and 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%
35.3	10	VALCOR ENGINEERING CORP	V526-5295-31	RC-241A, 262A and 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	6.66E+07	2.00E+08	> 50%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
35.5	10	TARGET ROCK	79AB-008	RC-241A, 262A and 291A	4.50E+07	1.80E+07	6.30E+07	2.16E+07	7.26E+07 Note 1	1.85E+08	> 50%
61.1	5	ROCKWELL INTERNATIONAL CORP. Note: NRC Safety Evaluation dated 3/22/2005 (ADAMS Accession No. ML050840156) determined that the Hydrogen Recombiners are not required and these components are currently being removed from the Dominion EQ program.	Hydrogen Recombiner 19690018-01B	RECOM-1	1.32E+03	1.00E+06	1.00E+06	1.20E+06	1.20E+06	1.80E+06	33%

Table III-5 (Continued)
MUR DOSE ASSESSMENT OF EQ COMPONENTS

NAPS QDR	QDR Rev.	MANUFACTURER	MODEL TYPE	EZD ZONE	Normal Dose (60 YR) [RADS]	Accident Dose Pre-MUR [RADS]	TID Dose Pre-MUR (Normal + Accident) [RADS]	Accident Dose Post-MUR (accident dose x 1.2) [RADS]	TID Dose Post-MUR [RADS]	Vendor's Qualified Dose [RADS]	Margin
71.1	18	DELPHI CONTROL SYSTEMS, INC Note: NRC Safety Evaluation dated 3/22/2005 (ADAMS Accession No. ML050840156) determined that the Hydrogen Monitoring Equipment is not required and these components are currently being removed from the Dominion EQ program.	Hydrogen Monitoring Equipment P1515 and other components	AB-259A, General	7.95E+03	1.00E+06	1.01E+06	1.20E+06	1.21E+06	1.1E+06	-10%

Note 1: For conservatism, beta dose indicated in QDR was added to Post-MUR TID.

EQ of electrical equipment is discussed in Section V.1.C, including disposition of equipment results for which a lower margin necessitated a refinement of the analysis as indicated in Table III-5.

III REFERENCES

III-1 EMF-2103(P)(A), Revision 0, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors,* AREVA Richland, Inc., April 2003.

III-2 Letter from Stephen Monarque (USNRC) to David A. Christian (Dominion), *North Anna Power Station, Unit 1 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel (TAC No. MB4714),* August 20, 2004.

III-3 Letter from Stephen Monarque (USNRC) to David A. Christian (Dominion), *North Anna Power Station, Unit 2 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel (TAC No. MB4715),* April 1, 2004.

III-4 Letter from Gerald T. Bischof (Dominion) to USNRC Document Control Desk, *Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, License Amendment Request Stretch Power Uprate,* Serial No. 07-0450, July 13, 2007.

III-5 Letter from Leslie N. Hartz (Dominion) to USNRC Document Control Desk, *North Anna Power Station Units 1 and 2, Supplemental Information for Realistic Large Break Loss of Coolant Accident (RLBLOCA) Containment Pressure Analysis, Proposed Technical Specification Changes and Exemption Request for Use of Framatome ANP Advanced Mark-BW Fuel,* Serial No. 03-313J, January 6, 2004.

IV. MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

1. A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.

   A. This discussion should address the following components:

      i. reactor vessel, nozzles and supports
      ii. reactor core support structures and vessel internals
      iii. control rod drive mechanisms
      iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles
      v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, containment systems)
      vi. steam generator tubes, secondary side internal support structures, shell, nozzles
      vii. reactor coolant pumps
      viii. pressurizer shell, nozzles, surge line
      ix. safety-related valves

   B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:

      i. stresses
      ii. cumulative usage factors (fatigue)
      iii. flow induced vibration
      iv. changes in temperature (pre- and post-uprate)
      v. changes in pressure (pre- and post-uprate)
      vi. changes in flow rates (pre-and post-uprate)
      vii. high energy line break locations
      viii. jet impingement and thrust forces

   C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel integrity with respect to:

      i. pressurized thermal shock calculations
      ii. fluence evaluation
      iii. heatup and cooldown pressure-temperature limit curves
      iv. low temperature overpressure protection
      v. upper shelf energy
      vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs, and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube cycle fatigue is consistent with NRC Bulletin 88-02, *Rapidly Propagating Fatigue Cracks in Steam Generator Tubes*, February 5, 1988.

RESPONSE TO IV - MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

IV.1.A.i Reactor Vessel

The reactor vessel stress and fatigue usage factors were evaluated at the uprated operating conditions. The evaluation assessed the effects of the revised operating parameters on the most limiting locations. The NAPS reactor vessels were originally analyzed with a normal operating inlet temperature of 543.0°F and a normal operating outlet temperature of 606.0°F. During a 1984 stretch power uprate, the analyzed normal operating temperatures were modified to a vessel inlet temperature of 552.3°F and vessel outlet temperature of 621.2°F, to agree with the design parameters. The MUR minimum vessel inlet temperature of 545.4°F and maximum vessel inlet temperature of 551.7°F are bounded by the current analysis (543.0°F–552.3°F) for those reactor vessel regions in contact with vessel inlet water. The MUR maximum vessel outlet temperature increased 0.7°F to 621.9°F.

The maximum vessel outlet temperature increase of 0.7°F increases the T_{hot} variations in the upper head and outlet nozzles during normal plant loading and unloading. Therefore, the evaluation considers the normal plant loading and plant unloading as slightly more severe transients at the main closure flange assembly, CRDM housings, and outlet nozzles. The North Anna Unit 1 and 2 reactor vessel closure heads were replaced in 2003. The stress analyses for the replacement heads were based on operating conditions which envelope the MUR operating conditions. Therefore, the main closure flange assembly and CRDM housing penetrations are fully acceptable for the MUR power uprate. The MUR transients for the outlet nozzles are slightly more severe, but remain bounded by the conservative design transients. The maximum cumulative usage factor does not change.

The remaining reactor vessel regions (including the inlet nozzles, vessel shell, and bottom head) are in contact with vessel inlet water during normal operation. Since the entire range of normal operating vessel inlet temperatures are bounded, no further evaluation was required for these regions.

The reactor vessel faulted condition stress analysis is unchanged by the MUR power uprate, because the previously analyzed faulted condition loads remain valid and below the allowable limits.

The code of record is listed in Section IV.1.D and remains unchanged. The reactor vessel meets the stress and fatigue analysis requirements of ASME B&PV Code, Section III, for plant operation at the uprated power conditions.

IV.1.A.ii Reactor Vessel Internals

The revised design conditions were evaluated for impact on the existing reactor vessel internals design basis analyses, as follows.

IV.1.A.ii.a Core Bypass Flow

The design core bypass flow limit is 6.5% of the total reactor vessel flow. This core bypass flow limit remains unchanged and valid for power uprate conditions. The MUR power uprate has an insignificant effect on the core bypass flow; core bypass flow remains below the 6.5% limit.

IV.1.A.ii.b Rod Control Cluster Assembly Drop Time

RCCA drop time is affected by changes to the RCCA driveline, fuel assembly hydraulic characteristics, and plant operating conditions. The power uprate does not change the RCCA driveline. The fuel assembly thermal-hydraulic characterization is not significantly changed. There is no impact on RCCA drop times due to the fuel thermal-hydraulic characteristics. The effect of increased core power with a decreased core inlet temperature of 0.4°F has an insignificant (less than 0.01 second increase) effect on RCCA drop time. The maximum measured RCCA drop time of 2.14 seconds at a single rod location, compares to 2.25 seconds used in the analysis. Based on these comparisons and the insignificant power uprate impact on expected RCCA drop times, the RCCA drop time results from previous analyses remain bounding.

IV.1.A.ii.c Hydraulic Lift Forces and Pressure Losses

An evaluation was performed to determine the hydraulic lift forces on the various reactor internal components, to ensure that the reactor internals assembly remains seated and stable for the applicable design conditions. The results indicate that the downward force remains essentially unchanged, and that reactor internals would remain seated and stable at the MUR power uprate conditions.

IV.1.A.ii.d Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or fuel rod vibration caused by a high-velocity water jet. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core. The baffle jetting phenomenon could lead to fuel cladding damage. There is no significant change to the pressure differential across the baffle plate, baffle gap width, and fuel assembly model response due to the power uprating. Therefore, the baffle joint momentum flux would not change as a result of the MUR power uprate.

IV.1.A.ii.e Mechanical Evaluation

The power uprated conditions do not affect the current design bases for seismic and LOCA loads. The flow induced vibration stress levels on the core barrel assembly and upper internals are low and below the material high-cycle fatigue endurance limit. Therefore, the MUR uprated conditions do not affect the structural margin for flow-induced vibration.

IV.1.A.ii.f Structural Evaluation

Evaluations were performed to demonstrate that the structural integrity of reactor components is not adversely affected by the MUR power uprate. For reactor internal components except the lower core plate and the upper core plate, the stresses and cumulative usage factor of the previous analyses remain bounding at power uprate conditions.

IV.1.A.ii.f.1 Lower Core Plate Structural Analysis

The lower core plate is subjected to the effects of heat generation rates, due to its proximity to the core. Structural evaluations were performed to demonstrate that the lower core plate structural integrity was not adversely affected by the revised design conditions. The lower core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, is acceptable. The lower core plate is structurally adequate for the MUR power uprate conditions.

IV.1.A.ii.f.2 Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel with installed baffle plates. Bolting connects former plates to the baffle and core barrel. This bolting restrains baffle plate motion. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, and secondary loads consisting of preload and thermal loads resulting from RCS temperatures and gamma heating rates.

An evaluation of the baffle former bolt maximum displacement was performed at MUR power uprate conditions. This displacement is caused by the temperature difference between the baffle and barrel regions, which is influenced by the power in the fuel assemblies adjacent to the baffle plates. The original analysis assumed that fresh fuel assemblies were loaded adjacent to the baffle. Power on the peripheral fuel assemblies is less than the initial power distribution, because only irradiated assemblies are loaded in the peripheral core locations. The core power distribution effect (lower power levels of peripheral fuel assemblies) offsets the increased loads due to gamma heating rates, resulting in a temperature difference less than the previous analysis of record. The power uprate baffle former bolt displacements are also less than the previous analysis of record. Therefore, the existing baffle-barrel region thermal and structural analysis results remain bounding for the MUR revised design conditions.

IV.1.A.ii.f.3 Upper Core Plate Structural Analysis

The maximum stress contributor in the upper core plate is the membrane stress resulting from the average temperature difference between the center portion of the upper core plate and the rim. The increased stress from increased gamma heating was determined as a function of heat generation rate increment. The fluid temperature effect resulting from the power uprate was small. The evaluation results indicate that the upper core plate structural integrity is maintained at power uprate conditions. The upper core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, is acceptable. The upper core plate is structurally adequate for the MUR power uprate conditions.

The reactor vessel internals evaluations conclude that these components continue to meet their design criteria at the MUR power uprate conditions.

IV.1.A.iii Control Rod Drive Mechanism

The CRDMs use electro-magnetic coils to position the RCCA within the reactor core. The updated design conditions (design parameters and NSSS design transients) were reviewed for impact on the existing CRDM design basis analyses. CRDMs are subjected to T_{hot} temperatures and RCS pressures. These are the only design parameters considered in the CRDM evaluation. The maximum T_{hot} from the uprated design parameters for any case is 621.9°F. The maximum T_{hot} of 622°F used in the analysis of record is bounding. No changes in RCS design or operating pressure were made as part of the power uprate. The hot leg temperature and pressure transients are unchanged from those used in the analysis of record. Therefore, the original transient analysis remains bounding and applicable to the uprated conditions. The stress intensity limits are based on the design temperature of 650°F and pressure of 2500 psia, which are unchanged by the power uprate. The only exception to this is the bell-mouthing evaluation for the upper joint threaded area, where the stress intensity limits are based on the local temperature obtained from analysis. The bell-mouthing evaluation at the upper joint was updated and shown acceptable for the power uprated conditions. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.iv Reactor Coolant Piping and Supports

The revised design conditions were evaluated for impact on the existing design basis analyses for the reactor coolant loop piping, including the loop bypass line and the pressurizer surge line, primary equipment nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge), primary equipment supports (reactor pressure vessel nozzle supports, SG columns, snubbers and lateral bumpers, and RCP columns and tie rods), reactor coolant loop branch nozzles (accumulator and charging line), and Class 1 auxiliary piping systems attached to the reactor coolant loop. There are no significant changes to the reactor coolant loop thermal analysis, LOCA analysis, main steam line break analysis, and reactor coolant loop piping system fatigue evaluations. The existing design transients remain valid for the uprated conditions. The T_{hot} and T_{cold} variations are conservative and bounding. There were no changes to any existing pressurizer design transient parameter responses. There are no significant changes to the pressurizer surge line operating conditions.

In conclusion, the current design basis reactor coolant loop piping system analysis remains applicable for the MUR power uprate conditions. There are no changes to the following: reactor coolant loop displacements at the Class 1 auxiliary line connections to the reactor coolant loop, Class 1 auxiliary lines, primary equipment nozzle qualification, branch nozzle qualification, and primary equipment supports loads. The maximum primary and secondary stresses and maximum usage factors remain valid. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.v Balance-of-Plant Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems and Containment Systems)

BOP piping includes NSSS interface systems, safety-related cooling water systems, and containment systems. The MUR uprate operating conditions for the BOP piping were reviewed for impact on the existing piping and supports design basis analyses.

Change factors were determined, as required, to evaluate and compare the changes in operating conditions. Thermal, pressure, and flow rate change factors were based on the following ratios.

- The thermal change factor was based on the ratio of power uprate to pre-uprate operating temperature. $(T_{UPRATE} - 70°F)/(T_{PRE\text{-}UPRATE} - 70°F)$

- The pressure change factor was determined by the ratio of ($P_{UPRATE}/P_{PRE\text{-}UPRATE}$)
- The flow rate change factor was determined by the ratio of ($Flow_{UPRATE}/Flow_{PRE\text{-}UPRATE}$)

These thermal, pressure and flow rate change factors were used in determining piping systems acceptability for power uprate conditions. When the change factors are less than or equal to 1.0 (the pre-uprate condition envelops or equals the power uprate condition), the piping system was considered acceptable for power uprate conditions. When the change factors are greater than 1.0, an evaluation was performed to address the specific temperature, pressure and/or flow rate increase to document piping system acceptability.

The BOP piping systems reviewed remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure and flow rate effects resulting from the MUR power uprate conditions.

IV.1.A.vi Steam Generator

The original Unit 1 and 2 Model 51 SGs were replaced in 1993 and 1995 respectively. The Model 54F replacement SGs are a blend of a new tube bundle, lower shell and primary channel head region, with the original upper shell (Model 51 steam drum) region. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.vi.1 Steam Generator Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation focused on changes to secondary side operating characteristics at MUR power uprate conditions. SG secondary side performance characteristics such as steam pressure and flow, circulation ratio, bundle mixture flow, heat flux, secondary side pressure drop, moisture carryover, hydrodynamic stability, secondary side mass and others are affected by increases in power level. Secondary side performance characteristics were calculated using the SG performance code GENF (secondary side characteristics except DNB). GENF code analyses were performed for the design parameter cases. A separate analysis was performed using the 3-D flow field analysis code ATHOS (DNB parameters) to determine the detailed flow parameters throughout the tube bundle. The thermal-hydraulic evaluation concluded that the NAPS SG thermal-hydraulic operating characteristics remain acceptable for the MUR power uprate.

IV.1.A.vi.2 Steam Generator Structural Integrity

The structural evaluation focused on the critical SG components as determined by the design basis analyses stress ratios and fatigue usages.

For primary side components (including the divider plate, tubesheet and shell junctions, tube-to-tubesheet weld, and tubes), the applicable scale factors were the ratios of the baseline condition primary-to-secondary side differential pressures to the uprated conditions differential pressures. The scale factor was applied conservatively to both the thermal and pressure stresses. For the secondary side components (including the feedwater nozzle and secondary manway bolts), the decrease in secondary side pressure at uprated conditions was the basis for determining the applicable scale factors. The stress increase resulting from the steam pressure reduction was calculated. The additional stress was then used in calculating the resulting fatigue usage changes for operation at MUR power uprate conditions. The scale factors were applied to the stresses listed in the reference stress reports. The scaled stresses were also considered in determining the stress ranges involving transients that originate from, or lead to, full power.

An analysis was performed to determine if the ASME B&PV Code limits on design primary-to-secondary ΔP are exceeded for any applicable transient at power uprate conditions. The analysis determined that the maximum primary to secondary side differential pressures during normal operating transients are 1515 psi and 1542 psi for high T_{avg} and low T_{avg} temperatures respectively. The maximum primary to secondary side differential pressure during upset condition transients is 1550 psi for both high T_{avg} and low T_{avg} temperatures. These values are below the applicable design pressure limits of 1600 psi and 1760 psi for normal and upset conditions respectively. Therefore, the ASME B&PV Code design pressure requirements are satisfied.

The primary-plus-secondary stress range for primary side and secondary side components was evaluated. The maximum range of primary-plus-secondary stress was compared with the corresponding $3S_m$ limit of the ASME B&PV Code. For those situations where the $3S_m$ limit was exceeded, a simplified elastic-plastic analysis was performed consistent with the original design basis analysis. The analyzed components, with the exception of the secondary manway bolts, meet the ASME B&PV Code limits. See Section IV.1.B.ii for additional details on the SG manway bolt cumulative usage factors.

IV.1.A.vi.3 Steam Generator Tube Bundle Integrity, Flow Induced Vibration and Wear

Tube Integrity

The NAPS Model 54F replacement SGs contain thermally treated Alloy 690TT tubing and ASME SA-240 tube support plates with broached quatrefoil holes. The quatrefoil tube hole configuration results in reduced potential for contaminant concentration at tube support plate intersections by reducing the crevice area. The first eight tube rows were heat treated after bending to relieve stresses. Hydraulic tube expansion in the tubesheet region results in reduced residual stresses compared to mechanical roll expansion and a more uniform expansion compared to explosively expanded tubes. Thermally treated Alloy 690 is highly resistant to stress corrosion cracking. The replacement SGs have exhibited no indications of corrosion related tube degradation after ten cycles in Unit 1 and eight cycles in Unit 2. Actual tube plugging levels are essentially 0% since the replacement SGs were installed. Two SG tubes have been plugged on Unit 1 and six SG tubes have been plugged on Unit 2. Seven of these tube plugs were installed due to NDE anomalies and one plug was required due to mechanical wear. No active systematic corrosion mechanisms have been identified in the NAPS SGs. During NAPS SG monitoring and operational assessments, only tube support plate wear was identified as an existing SG tube degradation mechanism for NAPS Units 1 and 2. Potential mechanisms such as anti-vibration bar wear, loose parts wear, outside diameter stress corrosion cracking, and pitting were absent in the NAPS Units 1 and 2 SGs, but are included in the inspection planning. To date, the NAPS Model 54F SGs have had no incidence of primary water stress corrosion cracking because of the Alloy 690TT tubes high resistance to primary water stress corrosion cracking.

Potential tube degradation mechanisms resulting from potential localized chemistry changes at the tube surfaces after the power uprate are outside diameter stress corrosion cracking and pitting. Based on laboratory and operating experience and current NAPS operating and maintenance practices, the power uprate will not produce excessive degradation due to those mechanisms. On the basis of T_{hot} temperature increase alone, the mechanical wear processes are unlikely to be significantly changed. The increased RCS temperature affects on primary water stress corrosion cracking are expected to be negligible. SG chemistry effects on tube corrosion after power uprating are insignificant.

Flow Induced Vibration and Wear

SG tube wear (i.e., fretting) was evaluated based on current design basis analysis and consideration of SG secondary side thermal-hydraulic changes resulting from the MUR power uprate. 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.

Predicted tube vibration amplitudes and fluid-elastic stability ratios are low, < 0.001 inch and 0.69, respectively. Wear in the straight leg region for design and postulated bounding conditions is < 0.008 inch. The analysis results indicate an increase in fluidelastic stability of 19%, with an increase in vibration amplitude due to turbulence and tube wear of 41%. The tube stability ratio will increase by 19% resulting in a stability ratio of 0.82, which is less than the 1.0 allowable and acceptable. Increasing the baseline vibration amplitude by 41% results in an amplitude of less than 0.0015 inch, which is well below the tube-to-tube gap of 0.350 inch. Therefore, the turbulence effects are acceptable. The maximum replacement SG baseline wear is small, 0.008 inch over the 60 years of plant operation. The revised wear is projected to increase from 0.008 inch (pre-uprate) to approximately 0.012 inch (post-uprate). At most, wear over the remaining SG life anywhere along the SG tubes as a result of flow effects is 60% of the 0.020 inch repair limit (40% through wall wear). These values are for a worst-case scenario and assume operation at the power uprate conditions occur over the entire SG life, which is conservative. This amount of tube wear will not significantly affect tube integrity, and is acceptable.

Other items reviewed were tube stress and fatigue. Tube stress resulting from flow induced vibration concerns after the MUR power uprate is approximately 0.3 ksi. This stress level is below the ASME stress limits and the fatigue endurance limit. Therefore, tube stresses are acceptable at MUR power uprate conditions, the flow induced vibration induced loading fatigue usage factor is negligible, and fatigue degradation from flow induced vibration is not anticipated.

IV.1.A.vi.4 Steam Generator Steam Drum Evaluation

The Model 54F replacement SGs are a blend of a new tube bundle, lower shell and primary channel head region, with the original upper shell (Model 51 steam drum) region. The original feedwater rings and moisture separation equipment were not scheduled for replacement. However, during the replacement SG outage on Unit 2, the original carbon steel feedwater rings in SG B and SG C were found degraded and were replaced. Other than these two feedwater rings, the remaining steam drum equipment on Units 1 and 2 at the time of SG replacement were original equipment with service dating back to June 1978 for Unit 1 and December 1980 for Unit 2.

FAC in the SG steam drum region depends on numerous factors, including material composition, fluid velocity and turbulence, and secondary side water chemistry. Operation at uprated plant conditions will increase feedwater flow rates in the SGs with the possibility of initiating or accelerating the FAC process within the steam drum regions. Remote visual inspections of NAPS Units 1 and 2 steam drum components have been performed to establish a baseline for comparing future inspections results. Minor FAC conditions were identified on the original SG feedwater ring components (excluding Unit 2 SGs B and C). However, no FAC was identified in the primary or secondary separators. Dominion will continue to perform steam drum component inspections after every two plant operating cycles to determine if the increased feedwater flow rates have initiated or accelerated the FAC process. Based on inspection results, the inspection frequency may be adjusted.

IV.1.A.vi.5 Steam Generator Mechanical Repair Hardware

Mechanical repair hardware refers to components such as plugs, sleeves, and stabilizers that are installed in the SGs to address tube degradation.

Analysis results showed that mechanical plug designs satisfy applicable stress, fatigue and retention acceptance criteria for operation at MUR power uprate conditions. There are no shop welded plugs in any NAPS replacement SG, so a shop welded plug evaluation was not required. There are no Alloy 600 tube plugs in any NAPS replacement SG and no Alloy 600 tube plugs will be installed in the future, so existing NRC rules on Alloy 600 tube plugs are not applicable. The NPT-80 field installed weld plug may be used in applications that cannot employ a mechanical plug. The NPT-80 weld plug remains qualified at power uprate conditions. Field machining SG tube ends is a possibility for modifications and tube repair (i.e., plugging, sleeving and tube end reopening). The analysis concluded that the revised stresses were within the ASME B&PV Code allowable values. The fatigue usage values, when adjusted for the power uprating, remained less than the 1.0 fatigue limit. Straight-leg sleeved cable stabilizers remain qualified for NAPS.

Therefore, SG repair hardware continues to meet ASME B&PV Code limits for plant operation at MUR power uprate conditions.

IV.1.A.vi.6 Steam Generator Loose Parts

There are no foreign objects present in Unit 1 SG-A or B or any Unit 2 SG. Foreign object search and retrieval operations during previous North Anna refueling outages determined that one unretrievable object (one inch long piece of wire from a wire brush) was present in the secondary side of Unit 1, SG-C after the 2001 outage. Since the actual object location is not known, the evaluation conservatively assumed the worst tube location with respect to tube wear potential in the SG tube bundle.

The previous loose part evaluation was reviewed to determine the power uprate effects on the object projected wear times. Although there was no indication of wear present on any tubes adjacent to the foreign object, the wear time analyses were performed assuming 20% initial tube wear on the limiting tube location. The SG secondary side conditions will change as a result of the MUR operating conditions, however, these changes do not affect the previous evaluation conclusions.

The analysis determined that the amount of time required for the limiting foreign object orientation to wear a tube down to a minimum allowable tube wall thickness under conservative secondary side conditions was greater than 3 years or 2 operational cycles.

Therefore, operation at the MUR power uprate conditions is acceptable with the existing SG loose parts.

IV.1.A.vi.7 Regulatory Guide 1.121 Analysis

NRC Regulatory Guide 1.121 describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by inservice inspection must be repaired or removed from service. The acceptable degradation level is called the repair limit.

The Regulatory Guide 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. SG tubing structural limits were determined by analysis, for an assumed uniform thinning degradation mode in both the axial and circumferential directions. The allowable stress limits were established using the ASME B&PV Code, Section III, 1986, Code Case N-20-3 minimum strength properties. The limiting stresses during normal operation (Level A) and upset (Level B) service conditions are the primary membrane stresses due to the primary-to-secondary pressure differential across the tube wall. The postulated accident condition loads for the faulted (Level D) service condition are the LOCA, steam line break and design basis earthquake (DBE).

The allowable tube repair limit is established by adjusting the structural limit per Regulatory Guide 1.121 to take into account uncertainties in eddy current measurement, and an operational allowance for continued tube degradation until the next scheduled inspection. Analyses have been performed to establish the structural limit for the tube straight-leg (free span) region for degradation over an unlimited axial extent, and for degradation over a limited axial extent at the tube support plate and anti-vibration bar intersections. The existing tube repair limit is unaffected by the MUR power uprate and remains valid at uprate conditions.

IV.1.A.vii Reactor Coolant Pumps and Reactor Coolant Pump Motors

Revised RCS conditions were reviewed for impact on the existing RCP design basis analyses. The NSSS design parameters considered in the RCP evaluation are vessel inlet temperatures and RCS pressure. The reactor vessel inlet temperature at the RCP discharge is considered instead of the SG outlet temperature at the RCP inlet because the vessel inlet temperature is slightly higher due to pump heat. No changes in RCS design or operating pressure were made as part of the MUR power uprate. The maximum vessel inlet temperature for any NSSS design parameters case is 551.7°F. This temperature is lower than the vessel inlet temperature of 552.3°F previously evaluated for the replacement SGs. Due to lower allowable design stress limits, higher temperatures are more limiting for RCP structural design qualification and the NSSS parameter change for the MUR power uprate is therefore conservative. The MUR power uprate conditions remain bounded by the original design conditions or previously evaluated conditions. The existing NSSS design transients conservatively reflect the parameter change during the transients. The T_{cold} variations as they presently exist in the component analyses are conservative and bounding.

The RCP motor limiting design parameter is the horsepower loading at continuous hot and cold operation. The previous NAPS assessment evaluated the RCP motor for continuous hot and cold operation and for loads on the thrust bearings and remains bounding and applicable for the revised RCS conditions. The RCP motors are acceptable for the loads calculated at MUR power uprate RCS conditions. The maximum pump horsepower is 6969 hp, which is less than the RCP motor nameplate rating of 7000 hp.

The revised RCS conditions are acceptable for the RCP with respect to ASME B&PV Code structural integrity. The code of record is listed in Section IV.1.D and remains unchanged. Therefore, the revised MUR power uprate conditions remain bounded by the original design conditions or previously evaluated

IV.1.A.viii Pressurizer Structural Evaluation

The MUR operating conditions were reviewed for impact on the existing pressurizer design basis analysis. The limiting pressurizer conditions occur when the RCS pressure is high and the RCS T_{hot} and T_{cold} are low. No changes were made in RCS design or operating pressure as part of the power uprate. The minimum T_{hot} and T_{cold} values from the design parameter cases were used in the pressurizer evaluation. At the normal operating pressure of 2250 psia, the revised T_{hot} and T_{cold} temperature differences for normal operation are bounded by the original analysis.

The NSSS design transients did not change and were enveloped by the existing design transients. Pressure fluctuations during the uprate transients are the same as the original evaluations. The maximum pressure within each load category (Normal, Upset, Faulted and Test) has not changed from the value used in the original evaluations. Thus, the uprate transients have no effect on the primary stress evaluations previously performed.

The NAPS pressurizer lower head was previously evaluated for insurge/outsurge transient effects related to both design transients and operational transients that were not considered in the original design. The revised design parameters were evaluated for their effect on the previous evaluation conclusions. The revised design parameters have an insignificant impact on the previous fatigue results and they remain valid.

Therefore, the pressurizer meets the stress/fatigue analysis requirements for plant operation at the MUR power uprate conditions. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.ix Safety Related Valves

The revised design conditions were reviewed for impact on the existing safety-related valves design basis analyses. No changes in RCS design or operating pressure were made as part of the power uprate. The evaluations concluded that the temperature changes due to the power uprate are bounded by those used in the existing analyses. Safety-related valves were reviewed within the applicable system (Section VI) and program (Section VII.6.E) evaluations. None of the safety-related valves required a change to their design or operation as a result of the power uprate.

IV.1.A.x Loop Stop Isolation Valves

The revised design conditions were reviewed for impact on the existing loop stop isolation valve design basis analyses. No changes in RCS design or operating pressure were made as part of the power uprate. The loop stop isolation valves are located in each RCS hot leg and cold leg. Higher temperatures are more limiting for the design qualification, so the hot leg valves were chosen to bound both applications. The maximum T_{hot} from any design parameters case is 621.9°F. This value is below the loop stop isolation valve design temperature of 650°F. Thus, the increased hot leg temperature is bounded by the original loop stop isolation valve evaluations. The existing NSSS design transients conservatively reflect the parameter change during the transients. In addition, the T_{hot} variations as they presently exist in the component analyses are conservative and bounding.

Therefore, the original loop stop isolation valve evaluations remain bounding and applicable to the design parameters and NSSS design transients at MUR power uprate conditions. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.B.i Stresses

The revised design conditions for the NSSS components and BOP piping (NSSS interface systems, safety-related cooling water systems and containment systems) were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and cumulative usage factors) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping and supports), IV.1.A.vi (steam generator), IV.1.A.vii (reactor coolant pumps and motors), IV.1.A.viii (pressurizer), IV.1.A.ix (safety-related valves), and IV.1.A.x (loop stop isolation valves). No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes are within design limits. The evaluations reviewed maximum stress intensities/stress ranges, with comparison to stress allowables, cumulative usage factors (for Class 1), and other special stress limits.

IV.1.B.ii Cumulative Usage Factors

The revised design conditions for the NSSS components, piping, and interface systems were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and cumulative usage factors) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping), IV.1.A.vi (steam generator), IV.1.A.vii (reactor coolant pumps and motors), IV.1.A.viii (pressurizer), IV.1.A.ix (safety-related valves), and IV.1.A.x (loop stop isolation valves).

For Class 1 components and piping, the stress analyses considered the impact on fatigue life. The cumulative usage factors were determined to be acceptable (< 1.0) for a 60-year plant life for each of the components except the steam generator secondary side manway bolts.

The SG manway bolts meet applicable fatigue limits under MUR uprated conditions for a (total) service period of approximately 45 years. Therefore, these bolts must be replaced prior to 45 years of service life to meet the fatigue usage limit of 1.0, unless additional analysis is performed to show that considering actual transient cycles, actual fatigue life experienced is less than calculated and the bolts will be acceptable for a longer service period. With the exception of the SG secondary side manway bolts, the cumulative usage factors for the other components remain less than 1.0 for the MUR power uprate conditions. No other changes are required due to stress levels or fatigue life considerations.

IV.1.B.iii Flow Induced Vibration

SG flow induced vibration is discussed in Section IV.1.A.vi.3.

IV.1.B.iv Temperature Effects

IV.1.B.iv.1 Changes in Temperature (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the NAPS power uprate. The operating temperature changes are shown in Attachment 1 Table 4.0-2. Specific calculation outputs include T_{hot} and T_{cold}. The current T_{avg} window has been maintained at 580.8°F–586.8°F. There is an approximate 1.3°F increase in temperature across the core (T_{hot} increases approximately 0.7°F and T_{cold} decreases approximately 0.6°F) from current operating conditions due to the power uprate. There is no change to the RCS average temperature limit in Technical Specification 3.4.1.

Changes in main steam and feedwater system temperatures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.iv.2 Evaluation of Potential for Thermal Stratification

NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*, addresses thermal stresses in piping attached to the RCS that cannot be isolated. The MUR power uprate temperature changes, when compared to current operation and evaluated using EPRI Material Reliability Program, MRP-146 (Reference IV-13), will not cause changes in the potential for cyclical thermal stratification, or in the predicted temperature profiles and cycling frequencies, that would require any different management approach to this issue from the existing Dominion programs. In addition, the RCS design flow rates are essentially the same as the power uprate values. Thus, the effects of swirl penetration will not change due to the MUR power uprate.

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, addresses surge line thermal stratification. Surge line thermal stratification is driven by the temperature difference between the RCS hot leg and the pressurizer. The current hot leg operating temperature will increase 0.7°F from the power uprate. A higher hot leg temperature lowers the temperature differential between the hot leg and pressurizer, which reduces the stratification effects. There are no significant changes to the surge line operating conditions and therefore no significant changes to the pressurizer stratification loading.

IV.1.B.v Changes in Pressure (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the NAPS power uprate. There will be no change in RCS operating pressure as a result of the MUR power uprate. The nominal operating pressure is 2250 psig (Attachment 1 Table 4.0-2). There is no change to the RCS pressure limit in Technical Specification 3.4.1.

Changes in main steam and feedwater system pressures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vi Changes in Flow Rates (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the NAPS power uprate. The mechanical design RCS flow is shown in Attachment 1 Table 4.0-2 and remains unchanged for the power uprate. There is no change to the RCS flow limit in Technical Specification 3.4.1.

Changes in main steam and feedwater system flow rates are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vii High Energy Line Break

IV.1.B.vii.1 High Energy Line Break Locations

A review was performed to determine the power uprate impact on HELB systems. MUR power uprate operating temperatures, pressures, and mass flow rates were compared to the analyzed conditions. The review concluded that overall, the total pipe stresses were not significantly impacted. Therefore, the MUR power uprate does not result in any new or revised pipe break locations, and the existing design basis for pipe break, jet impingement and pipe whip remains valid.

IV.1.B.vii.2 Leak Before Break Evaluation

The existing leak before break analyses justified eliminating large primary loop pipe rupture from the NAPS structural design basis (References IV-2 and IV-3). The applicable pipe loadings, normal operating pressure, and temperature parameters at power uprate conditions were used to evaluate leak before break. The leak before break acceptance criteria are based on NRC Standard Review Plan, Section 3.6.3. These criteria are satisfied for primary loop piping at power uprate conditions. The recommended margins are satisfied, and the existing analyses conclusions remain valid. Therefore, the dynamic effects of RCS primary loop piping breaks are not considered in the NAPS structural design basis at MUR power uprate conditions.

IV.1.B.viii LOCA Forces Including Jet Impingement and Thrust

A LOCA hydraulic forces analysis generates the hydraulic forcing functions and hydraulic loads that occur on RCS components due to a postulated LOCA. No changes in RCS design or operating pressure were made as part of the power uprate. LOCA hydraulic forces increase with lower temperatures, so they are predominantly influenced by T_{cold}. The full power minimum T_{cold} remains unchanged from that previously assumed. Therefore, the assessment for vessel/internals, loop, and steam generator LOCA hydraulic forcing functions remains valid for the power uprate design conditions. There are no changes to methodology, results, or margin of safety with respect to LOCA hydraulic forces as a result of the MUR uprate conditions.

IV.1.B.ix Seismic Qualification

NAPS safety-related structures, systems and components are designed for both seismic and dynamic events as described in NAPS UFSAR Sections 3.5 through 3.10. The MUR power uprate impact on mechanical and electrical equipment seismic qualification, and the dynamic effects associated with pipe whip and jet impingement forces was evaluated. The mechanical and electrical equipment reviewed included equipment associated with systems essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, and preventing the significant release of radioactive material to the environment.

The primary input motions due to the design basis earthquake are not affected by the MUR power uprate. Seismic design is not impacted, because seismic requirements remain unchanged. Therefore, the seismic qualification of essential equipment supports is unaffected.

The mechanical and electrical equipment seismic qualification review demonstrated that the equipment will continue to meet the current NAPS licensing basis with respect to the requirements of General Design Criteria-1, 2, 4, 14, 30; 10 CFR 50, Appendix B; and 10 CFR 100, Appendix A.

IV.1.C.i Pressurized Thermal Shock

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of reactor vessel beltline materials to failure during a PTS event, to ensure that adequate fracture toughness exists during reactor operation. 10 CFR 50.61 provides the requirements, methods of evaluation, and safety criteria for PTS assessments.

PTS screening calculations were performed for NAPS Units 1 and 2 reactor vessel beltline materials using the end of the current 60 year operating license (EOL) neutron fluence values. Dominion concluded that NAPS Units 1 and 2 reactor vessel beltline materials will continue to meet the 10 CFR 50.61 PTS screening criteria (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds). The limiting Unit 1 RT_{PTS} value of 190.9°F applies to the lower shell forging 90400/292332 (Reference IV-7). The limiting Unit 2 RT_{PTS} value of 227.7°F applies to lower shell forging 990533/297355 (Reference IV-7). These limiting materials are unchanged from those provided to the NRC in Reference IV-11.

The PTS screening calculations performed at the end of the current operating license result in RT_{PTS} values that are consistent with those documented in the vessel integrity analysis of record. The MUR power uprate has no impact on 10 CFR 50.61 compliance; the reactor vessel will remain within its PTS limits after the MUR power uprate.

IV.1.C.ii Fluence Evaluation

Fluence calculations were based on the NRC approved methodologies described in References IV-4 and IV-5. These methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190 (Reference IV-6). The evaluation complies with Regulatory Guide 1.190, because the acceptance criteria are derived directly from Regulatory Guide 1.190, Section 1.4.3. This section states that a vessel fluence uncertainty of 20% (one sigma, 1σ) is acceptable for RT_{PTS}

and RT_{NDT} determination. The NRC approved methodology used for the NAPS Units 1 and 2 fluence evaluations has been demonstrated to satisfy this criterion. The Regulatory Guide 1.190 specific requirements incorporated in this methodology are:

- The calculations use neutron transport cross-sections from the latest version of the valuated Nuclear Data File (ENDF/B-VI).
- A P5 expansion of the scattering cross-sections is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.
- An S16 order of angular quadrature is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.
- An uncertainty analysis that includes calculation comparisons with test and power reactor benchmarks and an analytical uncertainty study has been completed and documented in NRC approved topical reports. The transport calculations overall uncertainty was demonstrated to be 13% (one sigma, 1σ). This uncertainty level meets the Regulatory Guide 1.190 requirement of 20% (one sigma, 1σ).

<u>North Anna Unit 1</u>

The calculations for Cycles 1 through 19 (23.0 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

Cycles 1 through 5 – 2775 MWt
Cycle 6 – 2834 MWt
Cycles 7 through 19 – 2893 MWt

A previous power uprate from 2775 MWt to 2893 MWt occurred during Cycle 6. The power level listed above (2834 MWt) for Cycle 6 represents a burnup weighted average of 2775 MWt and 2893 MWt computed as follows:

Cycle	Power	Burnup
6a	2775	7853
6b	2893	7962
6	2834	15815

Cycle 20 projections were based on cycle 19 spatial power distribution and core power of 2893 MWt. Cycle 21 and beyond were based on a bounding uprated core power level of 2956 MWt and the uprate fuel cycle design.

North Anna Unit 2

The calculations for cycles 1 through 18 (21.6 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

Cycles 1 through 4 – 2775 MWt
Cycle 5 – 2863 MWt
Cycles 6 through 19 – 2893 MWt

A previous power uprate from 2775 MWt to 2893 MWt occurred during cycle 5. The power level listed above (2863 MWt) for cycle 5 represents a burnup weighted average of 2775 MWt and 2893 MWt computed as follows:

Cycle	Power	Burnup
5a	2775	4318
5b	2893	12939
5	2863	17257

Cycle 19 projections were based on Cycle 18 spatial power distribution and core power of 2893 MWt. Cycle 20 and beyond were based on a bounding uprated core power level of 2956 MWt and the uprate fuel cycle design.

The reactor vessel integrity database update provided to the NRC in Reference IV-7 is based on peak fast neutron fluence (E > 1.0 MeV) values for NAPS Units 1 and 2 reactor pressure vessels at the EOL. The peak reactor vessel inner surface fluence (E > 1.0 MeV) values used in the analysis of record for EOL and the MUR power uprate fluence for the same time period are shown in Table IV-1.

Table IV-1
Peak Reactor Vessel Inner Surface Fluence

Unit	Maximum Analysis of Record Fluence	Methodology	Years Exposed	MUR Maximum Fluence	Methodology	Years Exposed
1	5.90 E19 n/cm^2	References IV-7 and IV-12	50.3 EFPY	5.14 E19 n/cm^2	References IV-4 and IV-5	50.3 EFPY
2	5.91 E19 n/cm^2	References IV-7 and IV-12	52.3 EFPY	5.25 E19 n/cm^2	References IV-4 and IV-5	52.3 EFPY

The analysis of record maximum fluence values are conservative (higher in value) compared to those calculated for both MUR power uprate operation and license renewal at 2956 MWt core power starting at North Anna 1 Cycle 21 and North Anna 2 Cycle 20. Therefore, the analysis of record fluence values for reactor pressure vessel analyses are bounding for the power uprate and will be retained.

Given the uncertainties associated with the two NRC approved methodologies, both analyses would meet the 20% (one sigma, 1σ) Regulatory Guide 1.190 (Reference IV-6) requirement. Therefore, either calculation would be acceptable. Comparison to surveillance capsule measurements indicates that both calculations are slightly conservative.

IV.1.C.iii Heatup and Cooldown Pressure/Temperature Limit Curves

10 CFR 50, Appendix G provides fracture toughness requirements for ferritic low alloy steel or carbon steel materials in the reactor coolant system pressure boundary. It also includes the requirements on upper shelf energy values used for assessing the safety margins of reactor vessel materials against ductile tearing, and for calculating plant pressure-temperature (P-T) limits. These P-T limits are established to ensure the structural integrity of reactor coolant system pressure boundary ferritic components during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The current heatup and cooldown curves (Technical Specification Figures 3.4.3-1 and 3.4.3-2) are licensed through the first 50.3 EFPY Unit 1 and 52.3 EFPY Unit 2. RT_{NDT} calculations have been performed per Regulatory Guide 1.99, Revision 2 (Reference IV-8) for NAPS Units 1 and 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. As stated in Section IV.1.C.ii, the fluence methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190 (Reference IV-6). The most limiting 1/4-T RT_{NDT} value of 218.5°F bounds the EOL and MUR power uprate limiting material for both NAPS Units. The current heatup and cooldown curves and low temperature overpressure protection setpoints (Reference IV-11) are bounding through EOL with the MUR power uprate and do not require update, because the fluence values for reactor pressure vessel analyses are bounded by the existing analysis of record.

IV.1.C.iv Low Temperature Overpressure Protection

As described in Section IV.1.C.iii, the current low temperature overpressure protection setpoints are bounding through EOL with the MUR power uprate and do not require update, because the fluence values for reactor pressure vessel analyses are bounded by the existing analysis of record.

IV.1.C.v Effect on Upper Shelf Energy Calculation

Upper shelf toughness was evaluated to ensure compliance with 10 CFR 50, Appendix G. If the limiting reactor vessel beltline material's Charpy upper shelf energy is projected to fall below 50 ft-lb, an equivalent margins assessment must be performed. The limiting reactor vessel beltline materials for NAPS Unit 1 and Unit 2 are the lower shell forgings.

Three in-vessel surveillance capsules have been withdrawn to date from each North Anna Unit. As a validation of the NAPS Units 1 and 2 current analysis of neutron exposure, the measured reaction rates were used in conjunction with the current neutron spectra for each withdrawn capsule as input to the NRC approved least squares dosimetry evaluation methodology. For neutron fluence ($E > 1.0$ MeV), the adjusted to calculated ratios (A/C) span a range from 0.88 to 0.90 with an average A/C of 0.89 ± 1.0% (one sigma, 1σ) for Unit 1, and 0.89 to 0.91 with an average A/C of 0.90 ± 1.5% (one sigma, 1σ) for Unit 2 for the three capsule data set for each unit. These comparisons fall well within the ± 20% criterion specified in Regulatory Guide 1.190 (Reference IV-6), thus validating the current calculations applicability to the NAPS Units 1 and 2 reactor pressure vessels.

EOL Charpy upper shelf energy results are shown in Attachment 1 of Reference IV-7. This data was extracted from the integrity analysis of record, where the decrease in Charpy upper shelf energy due to peak EOL fluence at the 1/4-T location is calculated from Regulatory Guide 1.99, Revision 2, Figure 2 trend curves (Reference IV-8). The 1/4-T upper shelf energy values for NAPS Units 1 and 2 beltline materials meet the 50 ft-lb acceptance criteria of 10 CFR 50, Appendix G at the end of the current 60-year license period, including the MUR power uprate. This conclusion is based on fluence values for reactor pressure vessel analyses being bounded by the existing analysis of record.

IV.1.C.vi Surveillance Capsule Withdrawal Schedule

The reactor vessel material surveillance program provides a means for determining and monitoring the reactor vessel beltline material fracture toughness, to support analyses for ensuring the structural integrity of reactor vessel ferritic components.

A withdrawal schedule has been established to periodically remove surveillance capsules from each NAPS unit's reactor vessel, to monitor the reactor vessel materials condition under actual operating conditions. The schedules are consistent with ASTM E-185-82 (Reference IV-9) and based on the projected neutron fluence in the analysis of record. After a review of Reference IV-10, the surveillance capsule monitoring program requirements are satisfied through EOL, including the MUR power uprate. The Unit 1 final withdrawal is Refueling Outage 33 in 2030. The Unit 2 final withdrawal is Refueling Outage 32 in 2029. The evaluations have concluded that there are no changes necessary to the capsule withdrawal schedules in NAPS UFSAR Tables 5.4-2 and 5.4-3; the current capsule withdrawal schedules remain valid.

IV.1.D Codes of Record

Table IV-2
Codes of Record

Component	Code	Code Class	Edition and Addenda
Reactor Vessel[1]	ASME III	A	1968 Edition through Winter 1968 Addenda
CRDM	ASME III	A	1968 Edition through Winter 1969 Addenda
Steam Generator[2]			
Tube side	ASME III	A	1968 Edition through Winter 1968 Addenda
Shell side	ASME III	A[3]	1968 Edition through Winter 1968 Addenda
Pressurizer	ASME III	A	1968 Edition through Winter 1968 Addenda
Reactor Coolant System			
Piping and supports	ANSI B31.7 USAS B31.7	1	1969 Edition, and the 1970 and 1971 Addenda
Surge pipe[5]	ANSI B31.7 USAS B31.7	1	1969 Edition including 1970 and 1971 Addenda
Loop bypass line	ANSI B31.7 USAS B31.7	N/A	1969 Edition including 1970 and 1971 Addenda
Loop stop valves	ASME III	A	1968 Edition through Summer 1969 Addenda
Safety valves	ASME III	A	1968 Edition through Winter 1968 Addenda
Relief valves	ANSI B16.5	N/A	1968 Edition through Winter 1968 Addenda
Reactor coolant pump	ASME III	A	1968 Edition through Winter 1970 Addenda
Main Steam System			
Piping	ANSI B31.1.0[4]		1967 Edition
Safety Valves (MSSV)	ASME III		1968 Edition through Winter 1970 Addenda

Table IV-2 (Continued)
Codes of Record

Component	Code	Code Class	Edition and Addenda

1. The reactor vessel closure heads were fabricated and manufactured in accordance with the French Construction Code (R-CCM) 1993 Edition with, 1st Addenda June 1994, 2nd Addenda June 1995, 3rd Addenda June 1996 and modification sheets FM 797, 798, 801 through 807. The sizing calculations and the stress and fatigue analysis were performed to ASME B&PV Code, Section III, 1995 Edition 1996 Addenda. The Design Reports certified that the closure heads meet the design requirements and stress limits for the ASME B&PV Code, Section III, 1968 Edition through Winter 1968 Addenda.
2. Code edition is for Class I Stress Reports. Replacement steam generators were fabricated and manufactured in accordance with the 1986 Edition of ASME III.
3. Code design requirements assigned are in excess of the requirement dictated by the applicable Safety Class.
4. Except the main steam piping designated as Seismic Class I, which is designed per ANSI B31.7.
5. Surge line was evaluated later using 1986 version of ASME Section III Code to address thermal stratification issue to meet the requirements of the NRC Bulletin 88-11.

There are no changes to the codes of record listed above in Table IV-2.

IV.1.E Changes to Component Inspection and Testing Programs

IV.1.E.i Inservice Testing Program

10 CFR 50.55a(f), *Inservice Testing Requirements*, mandates the development and implementation of an IST Program. NAPS has developed and is implementing an IST Program for pumps and valves per the applicable requirements. NAPS Technical Specification 5.5.7 describes the surveillance requirements that apply to the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

The applicable system analyses were reviewed to determine if the MUR power uprate would impact the existing IST Program. There are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect component performance or test acceptable criteria. Therefore, the MUR power uprate has no impact on the testing required by the IST Program.

IV.1.E.ii Inservice Inspection Program

10 CFR 50.55a(g), *Inservice Inspection Requirements,* mandates the development and implementation of an ISI Program. The applicable program requirements are specified in ASME B&PV Code, Section XI. NAPS has developed and is implementing an ISI Program per these requirements. UFSAR Section 5.2.5 describes the ISI Program.

This evaluation reviewed the MUR power uprate impact on the existing ISI Program. System classifications or boundaries for ASME Class 1, 2, and 3 systems are not affected. Inspection frequencies and required procedures for ASME Class 1, 2, and 3 components and their supports as described in the ISI Program Manual are not affected. Therefore, the MUR power uprate has no impact on the existing ISI Program.

IV.1.E.iii Erosion/Corrosion Program

NAPS has established and maintains a FAC Program per NRC Generic Letter 89-09, Erosion/Corrosion – Induced Pipe Wall Thinning. The FAC Program meets the intent of EPRI NSAC-202L, Recommendations for an Effective Flow-Accelerated Corrosion Program, and INPO EPG-06, INPO Engineering Guide – Flow Accelerated Corrosion. This program provides a standardized method of identifying, inspecting, and tracking components susceptible to FAC wear in both single and two-phase flow conditions. Program elements include: FAC susceptibility analysis and modeling, FAC inspection and evaluation, operational experience reviews, and crossover/crossunder main steam piping and moisture separators/reheaters inspections and evaluations. In general, plant systems are considered susceptible to FAC unless excluded by defined criteria. The criteria includes: material, moisture content, temperature, dissolved oxygen, frequency of system usage, plant-specific operating experience, and industry operating experience. NAPS utilizes the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code to predict and track FAC susceptible components. The CHECWORKS SFA computer code has been used to create unit-specific databases. Once the database has been built, the application is used to perform analysis and data interpretation. These analytical models result in Wear Rate Analysis that rank components in order of predicted FAC wear and predicted time to reach minimum code wall thickness. In order to evaluate the power uprate impact on FAC wear rates, the NAPS Unit 1 and 2 CHECKWORKS SFA models were updated to incorporate the changes associated with the power uprate.

NAPS Unit 1and 2 evaluations were performed to determine the impact on remaining service life as a result of the increase in wear rates due to the MUR power uprate. Tables IV-3 and IV-4 summarize these reviews.

Table IV-3
North Anna Unit 1 Wear Rate Analysis

Model	System	Increase in Wear Rate	Decrease in Time to T_{crit} (code wall)	Notes
FW Main 901#	Feedwater	1.3%	1.4%	Exceeds remaining plant life.
Drain of 3A & 3B	Steam Drain	6.2%	6.0%	Exceeds remaining plant life.
HP Drain Pump Suction	Steam Drain	10.4%	9.4%	Exceeds remaining plant life.
MSR Drain Lines	Steam Drain	14.9%	9.0%	Exceeds remaining plant life.
Drain of 6A & 6B	Steam Drain	16.9%	14.6%	Exceeds remaining plant life.

Table IV-4
North Anna Unit 2 Wear Rate Analysis

Model	System	Increase in Wear Rate	Decrease in Time to T_{crit} (code wall)	Notes
FW Htr 901 Inlet	Feedwater	**5.3%**	**5.13%**	Exceeds remaining plant life.
MSR Drain Lines	Steam Drain	**5.6%**	**5.38%**	Exceeds remaining plant life.
HP Drain Pump Discharge	Steam Drain	**5.7%**	**5.24%**	Exceeds remaining plant life.
HP Drain Pump Suction	Steam Drain	**6.0%**	**5.65%**	Exceeds remaining plant life.
ES #2 pt	Extraction Steam	**9.5%**	**N/A**	Piping is FAC Resistant

Tables IV-3 and IV-4 represent randomly selected piping components in the five systems expected to experience the greatest increase in FAC wear as a result of the MUR power uprate. The randomly selected piping components in other systems have a smaller increase or an actual decrease in FAC wear.

Upon power uprate implementation, the CHECKWORKS SFA databases for NAPS Units 1 and 2 will be updated and validated. The wear rate analysis models will be analyzed using the updated information and the Wear Rate Analysis – Service Life Report for each model will be reviewed. Any piping components with a low or a negative time for remaining service life will be evaluated for re-inspection.

Based on the reviews conducted for the impact of increased wear rates on remaining service life, there is no significant impact. No additional secondary system lines were identified as requiring monitoring for FAC wear as a result of the MUR power uprate. The remaining service life for the modeled FAC susceptible lines will continue to be monitored and will be documented at the end of each refueling outage.

IV.1.F Impact of NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

NRC Bulletin 88-02 required actions by operating license holders of Westinghouse designed nuclear power reactors with SGs having carbon steel support plates. SGs in this category include Westinghouse models 13, 27, 44, 51, D1, D2, D3, D4 and E. These actions were required to minimize the potential for a steam generator tube rupture caused by rapidly propagating fatigue cracks such as occurred at North Anna 1 on July 15, 1987. The tube rupture was caused by high cycle fatigue.

As previously stated, NAPS Units 1 and 2 Model 51 SGs were replaced in 1993 and 1995, respectively. The Model 54F replacement SGs are a blend of a new tube bundle, lower shell and primary channel head region, with the original upper shell (Model 51 steam drum) region. An evaluation was performed on the potential for high cycle fatigue in unsupported SG U-bend tubes. One of the prerequisites for high cycle SG U-bend fatigue is a dented support condition at the upper plate. This support condition results from corrosion product build-up associated with drilled holes in carbon steel tube support plates. Since the broached stainless steel support plate in this model SG is designed to inhibit the introduction of corrosion products, the support condition (i.e., denting) necessary for high cycle fatigue should not occur. Dominion has not observed any corrosion product build-up to date. Therefore, high cycle fatigue associated with unsupported inner row SG tubes is not a concern in this model SG.

IV REFERENCES

IV-1 NEI-97-06, Revision 2, *Steam Generator Program Guidelines*, May 2005.

IV-2 WCAP-11163, *Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for North Anna Units 1 and 2*, August 1986.

IV-3 WCAP-11163 Supplement 1, *Additional Information in Support of the Technical Justification for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for North Anna Units 1 and 2*, August 1988.

IV-4 WCAP-14040-NP-A, Revision 4, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, May 2004.

IV-5 WCAP-16083-NP-A, Revision 0, *Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry*, May 2006.

IV-6 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March 2001.

IV-7 Letter from Leslie N. Hartz (Dominion) to USNRC Document Control Desk, North Anna Power Station Units 1 and 2, *Update to Reactor Vessel Integrity Database to Reflect License Renewal Period*, Serial No. 05-834, December 13, 2005.

IV-8 Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*, May 1988.

IV-9 American Society for Testing and Materials (ASTM) E185-82,*Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.*

IV-10 Letter from E.C. Marinos (NRC) to David Christian (Dominion), *North Anna Power Station, Unit Nos. 1 and 2 – Approval of Proposed Reactor Vessel Material Surveillance Capsule Withdrawal Schedule (TAC Nos. MC6412 and MC6413),* March 15, 2006.

IV-11 Letter from W.R. Matthews (Dominion) to USNRC, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Change Request, Reactor Coolant System Pressure/Temperature Limits, LTOPs Setpoints and LTOPs Enable Temperatures,* Serial No. 04-380, July 1, 2004; supplemented by letters October 28, 2004 and November 16, 2004.

IV-12 Dominion Topical Report VEP-NAF-3-A, *Reactor Vessel Fluence Analysis Methodology*, April 1999.

IV-13 EPRI Material Reliability Program (MRP)–146,*Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant Branch Lines.*

V. ELECTRICAL EQUIPMENT DESIGN

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

   A. emergency diesel generators

   B. station blackout equipment

   C. environmental qualification of electrical equipment

   D. grid stability

RESPONSE TO V - ELECTRICAL EQUIPMENT DESIGN

V.1.A Emergency Diesel Generators

The Emergency Diesel Generator (EDG) system provides a safety-related emergency source of AC power for the engineered safeguards and selected BOP emergency loads, in the event that the normal AC power is interrupted. There are two EDGs for each NAPS unit dedicated to the safety-related, redundant electrical buses.

The electrical loads that changed as a result of the power uprate are not fed from the EDG system. There are no increases to the emergency buses loads supported by the EDGs. The EDG system equipment capacity and capability for plant operation at the uprate conditions are bounded by the EDG loading tables. The EDG loading tables are supported by the existing analysis of record. Both the bounding analysis and the EDG loading tables demonstrate that the EDG system has adequate capacity and capability to provide onsite standby power for safety-related loads following a loss of offsite power with or without a concurrent accident. Therefore, the EDG system is not affected by the MUR power uprate.

V.1.B Station Blackout Program

10 CFR 50.63 requires each light water cooled nuclear power plant to withstand and recover from a loss of all AC power, referred to as Station Blackout (SBO). The NAPS coping duration is four hours. This is based on an evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability. The evaluation was completed per NUMARC 87-00 and NRC Regulatory Guide 1.155. The MUR power uprate has no impact on the current

SBO coping duration of four hours. The MUR power uprate was evaluated for impact on the alternate AC power source and the following SBO coping issues: emergency condensate storage tank inventory, Class 1E battery capacity, ventilation, compressed air, and containment isolation.

V.1.B.i Alternate AC Power Source

The alternate AC power source consists of a diesel generator and support subsystems (e.g., starting air, cooling water, lubrication and fuel oil). The alternate AC diesel generator, with its separate fuel supply, can be aligned to any of NAPS four emergency buses (two per unit). This provides additional assurance that AC power will remain available. The alternate AC diesel generator has sufficient capacity to operate systems necessary for coping with a SBO event for the required coping period.

V.1.B.ii Emergency Condensate Storage Tank Inventory

The ECST provides adequate inventory to maintain a NAPS unit in hot standby for eight hours at MUR power uprate conditions. Since NAPS has a four-hour SBO coping period, the ECST provides adequate inventory for decay heat removal following a SBO event at uprated conditions. The SBO analysis assumes 2951 MWt, which is 102% of 2893 MWt.

V.1.B.iii Class 1E Battery Capacity

The NAPS Class 1E batteries have sufficient capacity to meet the SBO loads for one hour. Using the alternate AC power source, two battery chargers will be available within one hour on the blacked-out unit. The MUR power uprate does not affect any DC powered indication, control, or protection equipment. Therefore, the Class 1E batteries are acceptable at MUR power uprate conditions.

V.1.B.iv Ventilation

Evaluations have been performed for the following areas containing SBO equipment: turbine driven auxiliary feedwater pump room, charging pump cubicles, control room, emergency switchgear rooms, and containment. The turbine driven auxiliary feedwater pump room, charging pump cubicles, control room, and emergency switchgear rooms are unaffected by the MUR power uprate. The containment pressure and temperature resulting from a LOCA or MSLB envelope the SBO event at MUR power uprate conditions.

V.1.B.v Compressed Air

The power uprate does not affect the capability for manual operation of air-operated valves, or the capability to restore compressed air by powering an instrument air compressor immediately from the unaffected unit or within one hour on the SBO unit.

V.1.B.vi Containment Isolation

The power uprate does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is not related to power level. The evaluation for containment isolation at current plant conditions remains applicable at MUR power uprate conditions.

V.1.C Environmental Qualification of Electrical Equipment

The term EQ applies to equipment important-to-safety. The intent is to ensure this equipment remains functional during and following design basis events. The NAPS EQ Program has been developed to ensure that EQ criteria are applied to electrical equipment important to safety as specified in 10 CFR 50.49, and to document the process used to demonstrate this qualification. North Anna is licensed to implement the 10 CFR 50.49 requirements as follows:

Unit 1 NRC Bulletin 79-01 (Reference V-1) and IEEE Standard 323-1974 (Reference V-2)

Unit 2 NRC NUREG-0588 (Reference V-3) and IEEE Standard 323-1974 (Reference V-2)

There is no effect on EQ relative non-radiological conditions (e.g., temperature, pressure, humidity) resulting from the MUR power uprate. The source terms used for the radiation aspects of the EQ program evaluations have been adjusted as described in Section III.2 to accommodate MUR power uprate operation.

Radiation dose qualification is based on the sum of the normal operational dose plus the accident dose. The increase in the post-accident integrated dose conservatively determined for the power uprate (refer to Section III.2) has been evaluated for all equipment in affected environmental zones. The evaluation in Section III.2 (summarized in Table III-5) indicates that the increased radiation levels in some zones may impact equipment qualification for certain classes of equipment. Disposition of these specific cases is presented below.

<u>Excore Neutron Detectors</u>

After the current Spring 2009 Unit 1 outage, the excore neutron detectors will have been replaced with new equipment at both Unit 1 and 2. The Unit 2 excore neutron detectors were replaced in 2008. The equipment replacement was the result of existing EQ Periodic Maintenance schedules. Preliminary evaluation of results for the radiation analysis discussed in Section III.2.A indicate that the qualified installed lifetime will be greater than that for the original equipment currently maintained in the EQ program. Prior to operating above the current RTP of 2893 MWt, Dominion will incorporate changes in the qualified lifetime of this equipment into EQ program documentation.

Hydrogen Recombiners And Hydrogen Monitoring Equipment

A NRC Safety Evaluation was included in Amendment Nos. 238 and 219 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2, dated 03/22/2005 (ADAMS Accession No. ML050840156). This Safety Evaluation determined that the Hydrogen Recombiners are not required to be included in Station Technical Specifications and. The NRC has deleted the requirement for these components from 10 CFR 50.44. This safety evaluation also determined that the Hydrogen Monitoring Equipment no longer meets the definition of a safety-related component as defined in 10 CFR 50.2. Accordingly, these components are currently being removed from the Dominion EQ program as documented by CR323349.

Dominion has reviewed the effects of the proposed power uprate on the EQ of electrical equipment, and concludes that the evaluation has adequately addressed the effects of the proposed power uprate on the environmental conditions for the qualification of electrical equipment. Based on this evaluation, the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed power uprate. The impact of the one potential change for excore detectors will be resolved prior to MUR power uprate implementation, including any required changes in EQ program documentation for this equipment. Therefore, Dominion finds the proposed MUR power uprate acceptable with respect to the EQ of electrical equipment.

V.1.D Grid Stability

V.1.D.i Background

NAPS currently has a MVAR output limitation due to the 4 kV station service buses. The station service buses have a maximum voltage of 4.4 kV. The current generators are not capable of putting out their full MVAR capability at normal system voltage. The generators are capable of producing approximately 400 MVARs. However, because station service bus has a maximum voltage of 4.4 kV, the generator output is limited to approximately 200 MVARs. Dominion assessed the impact of a 170 MWe (i.e., 85 MWe per unit), of new generation capacity on the Dominion transmission system. Dominion is anticipating additional plant modifications that would result in additional electrical power increases beyond that proposed by this MUR LAR. Grid stability studies were conducted assuming that power increases were in effect, so the results bound the MUR power uprate. The transmission system assessment was based on Pennsylvania, New Jersey, Maryland Interconnection's (PJM) best assumptions at the present time for load growth and new generation through the summer of 2012. The evaluation included load flow studies of import/export system conditions and single-contingency, both normal and stressed, system conditions. Short circuit duty screening and stability analysis were also performed. Dominion considers a transmission facility overloaded if it exceeds 94% of its emergency rating under normal and stressed conditions.

V.1.D.ii Proposed New Generation Impact Analysis

Dominion routinely evaluates the impact that a proposed new generation resource will have under maximum generation conditions and stressed system conditions. Two different assessments were conducted: local generation and import/export conditions.

The local generation study assessed station operation at maximum capability. The study identified no transmission deficiencies. The import/export study assessed conditions into and out of the Dominion system. Any new facility interconnected with the Dominion system should not significantly decrement First Contingency Incremental Transfer Capability between utilities. The study indicated no decrement to system First Contingency Incremental Transfer Capability. In the summary section of the PJM system impact studies, the maximum facility output is 945 MWe for Unit 1 and 938 MWe for Unit 2. The MUR power uprate will increase each unit's generating capacity by approximately 15 MWe.

V.1.D.iii Stability Analysis

The range of contingenies evaluated was limited to that necessary to assess compliance with the Dominion criteria. Simulation time was limited to 25 seconds for faults. Two types of faults were considered in this study: three-phase faults with primary clearing time and stuck breaker fault followed by another single line to ground fault. No secondary protection faults were tested due to the presence of dual primary relays in area of study.

No transient stability issues related to the NAPS power uprate were identified. Therefore, the current grid configuration and capacity is adequate to handle the additional megawatts generated from the MUR power uprate. The details supporting the system stability for NAPS are contained in the PJM Generator Impact Study. The study contains the system impacts, power flow studies, network conditions, and supporting one-line diagrams.

V.1.E Onsite Power Systems

The AC Distribution System is the source of power for the non safety-related buses and the safety-related emergency buses. It consists of the 4.16 kV, 480 V, and 120 V systems (excluding the EDGs). The electrical changes resulting from the MUR power uprate occur in the NAPS equipment, primarily at the 4.16 kV voltage level. The following loads were affected by the uprate: main feedwater pump, condensate pump, LP heater drain pump, HP heater drain pump, and

RCP. None of the revised brake hp values exceeded the motor nameplate rating, but the operating points changed. An evaluation determined that current loading levels under MUR power uprate conditions are bounded by the 4.16 kV buses existing capability. There were no load increases on the 480 V buses. The 120 V system loads are not related to the power generation process and are, therefore, independent of the MUR power uprate. The 125 VDC system loads are also not related to the power generation process and are therefore independent of the MUR power uprate. Therefore, the AC 4.16 kV, 480 V, 120 V and DC 125 V electrical distribution systems are acceptable at MUR power uprate conditions.

V.1.F Power Conversion Systems

As a result of the MUR power uprate, the RTP will increase from 2893 MWt to 2940 MWt. This increase in thermal power will result in an increase in electrical power output, which affects power block equipment.

V.1.F.i Main Generator

Unit 1

The nameplate rating is 1105 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 475 MVARs out or 390 MVARs in, and maintain generator load and hydrogen pressure within the limits of the Generator Calculated Capability Curve with a generator rating of 1088.6 MVA. The main generator output at the current NSSS power level of 2905 MWt is 965 MWe. The anticipated main generator output is 980.5 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 980.5 MWe, the generator is capable of exporting 472.9 MVAR (lagging power factor of 0.900) and importing approximately 390 MVAR (leading power factor of 0.929). The exciter has the capability to support main generator operation within its restricted operational rating and within the capability curve for leading and lagging power factor. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability.

Unit 2

The Unit 2 main generator was replaced during the September 2008 outage. The exciter and voltage regulator were not replaced. The new nameplate rating is 1200 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 444 MVARs out or 210 MVARs in, and maintain generator load and hydrogen pressure within the limits of Generator Calculated Capability Curve with a generator rating of 1088.6 MVA. The main generator output at the current NSSS power level of 2905 MWt is 965 MWe. The anticipated main generator output is 972.9 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 972.9 MWe, the generator is capable of exporting approximately 444 MVAR (lagging power factor of 0.910) and importing approximately 210 MVAR (leading power factor of 0.977). The exciter has the capability to support main generator operation within its restricted operational rating and within the capability curve for leading and lagging power factor. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability.

V.1.F.ii Isolated Phase Bus

The isophase bus duct is rated for 30,500 amperes. The MUR power uprate will raise the isophase bus duct current to 30,072 amperes for Unit 1 and 29,862 amperes for Unit 2. Therefore, the increase from the MUR power uprate remains below the isophase bus maximum capability.

V.1.F.iii Main Generator Breaker (Unit 1 only)

A main generator circuit breaker has been installed on Unit 1. This breaker permits the normal station service transformers to supply the normal station service buses when the main generator is offline. This arrangement reduces the likelihood of simultaneously loading the normal and emergency buses on the reserve station service transformers from both Units 1 and 2. There are no plans to install a main generator circuit breaker on Unit 2. The main generator circuit breaker has a continuous current rating of 36,000 amperes. This continuous current rating has margin to the maximum generator output at MUR power uprate conditions.

V.1.F.iv Main (Step-up) Transformers

The main transformers increase the main generator 22 kV output voltage to the 500 kV transmission voltage. These transformers are rated for 1200 MVA, which is above the main generator 1088.6 MVA output capability. The transformers are sized to handle the MUR power uprate conditions.

V.1.F.v Unit Station Service Transformers

The unit station service transformers are supplied by the 22 kV isolated phase bus and power the 4.16 kV switchgear, 480 V load centers and motor control centers during normal operating conditions. The 4.16 kV normal switchgear buses are transferred and connected directly to the secondary of the reserve station service transformers during station startup and shutdown conditions. The BOP electrical loads affected by the uprate increase the loading on the unit station service transformers. Even with the increased load, the unit station service transformers remain within their current rating.

V.1.F.vi Reserve Station Service Transformers

The reserve station service transformers are supplied by the 34.5 kV switchyard and 4.16 kV transfer buses. The BOP electrical loads affected by the uprate increase the loading on the reserve station service transformers. Even with the increased load, the reserve station service transformers remain within their current rating.

V.1.G Switchyard

The current to the switchyard is bounded by the main transformers capability. The overhead lines from the main transformers to the switchyard are capable of carrying the full transformer load. Therefore, the overhead lines are acceptable at the MUR conditions. An evaluation determined that the small increase in power output does not significantly impact the switchyard equipment. The switchyard system analyses bound the MUR power uprate conditions.

V REFERENCES

V-1 NRC, Bulletin 79-01, *Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors.*

V-2 IEEE Standard 323-1974, *Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.*

V-3 NRC NUREG-0588, *Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment.*

VI. SYSTEM DESIGN

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems.

   A. NSSS interface systems for pressurized water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling water reactors (BWRs) (e.g., suppression pool cooling), as applicable

   B. containment systems

   C. safety-related cooling water systems

   D. spent fuel pool storage and cooling systems

   E. radioactive waste systems

   F. engineered safety features (ESF) heating, ventilation and air conditioning

RESPONSE TO VI - SYSTEM DESIGN

VI.1.A Interface Systems

VI.1.A.i Main Steam System

The main steam system is described in UFSAR Section 10.3. This system was evaluated to determine the impact of the MUR power uprate. Component parameters are bounded by the original design equipment ratings, or by the original design considerations for off-normal operation. Therefore, the main steam system is acceptable at power uprate conditions.

VI.1.A.i.a Main Steam Piping

Main steam system pressures, temperatures and velocities were evaluated. System pressures and temperatures are bounded by piping design parameters during power uprate conditions. The velocities were bounded by the maximum recommended velocities, with the exception of a short section of pipe feeding the low pressure turbines. This section of piping is included in the NAPS FAC

Program and will be monitored to ensure minimum wall is maintained. Main steam system piping is acceptable at MUR power uprate conditions.

VI.1.A.i.b Main Steam Safety Valves

A total of five ASME B&PV Code MSSVs are located on each main steam line outside reactor containment and upstream of the main steam trip valves (MSTVs). MSSV lift setpoints are determined by SG design pressure and the ASME B&PV Code. The SG design pressure has not changed with the MUR power uprate, so the existing MSSV setpoints are unchanged. Main steam overpressure events have been analyzed at 2951 MWt (102% of 2893 MWt) and the MSSVs are adequate for the MUR power uprate.

VI.1.A.i.c Main Steam Trip Valves and Non-Return Valves

The MSTVs provide a means to isolate a SG in the event of a downstream steam line rupture. The non-return valves are located downstream of the MSTVs and prevent reverse flow in the main steam lines. The MSTVs are required to close within five seconds in the event of a main steam line break. The power uprate does not affect the MSTVs' ability to close within the required time period. Design loads and associated stresses resulting from rapid valve closure do not change with the power uprate. The MUR power uprate steam flow is bounded by the maximum steam flow for the non-return valves. The worst case for differential pressure increase is controlled by the steam line break areas, SG flow restrictor throat area, valve seat bore, and no load operating pressure. Since the power uprate does not impact these variables, the maximum pressure design loads and associated stresses resulting from MSTV and non-return valve rapid closure will not change. The maximum differential pressure requirement remains satisfied. Therefore, the MSTVs and non-return valves are acceptable at MUR power uprate conditions.

VI.1.A.i.d Moisture Separator Reheaters

Shell side and tube side pressures remain bounded by the moisture separator reheater (MSR) design conditions at power uprate conditions. The MSR safety valves are acceptable at MUR power uprate conditions.

VI.1.A.ii Steam Dump

The NAPS steam dump function is accomplished by the SG PORVs (atmospheric relief valves) and the steam dump system (turbine bypass valves). The SG PORVs are described in UFSAR Section 10.3. The steam dump system is described in UFSAR Sections 7.7 and 10.3.

VI.1.A.ii.a Steam Generator PORVs

There are three SG PORVs per unit, one on each MS line. The SG PORVs are located upstream of the MSTV and adjacent to the MSSV. There is no change in function associated with the power uprate. The SG PORVs automatically modulate open and exhaust to the atmosphere whenever the steam line pressure exceeds a predetermined setpoint. This minimizes safety valve lifting during steam pressure transients. The SG PORV set pressure for these operations is between 0-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures change for the proposed range of NSSS operating parameters, the SG PORV setpoint is unchanged.

The primary function of the SG PORVs is to provide a means for decay heat removal and plant cooldown when the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. The SG PORVs are sized to have a capacity equal to approximately 10% of rated steam flow at no-load pressure. The SG PORVs have a capacity of 9.48% at uprated conditions. An evaluation of the installed capacity concluded that the original design bases, in terms of plant cooldown capability, can still be achieved for the range of power uprate NSSS design parameters. Therefore, the SG PORVs are acceptable for operation at uprate conditions.

VI.1.A.ii.b Steam Dump System

The steam dump system creates an artificial steam load by dumping steam to the main condenser. Each North Anna unit is provided with 8 condenser steam dump valves. Steam dump in conjunction with the reactor control system permits the NSSS to withstand an external load reduction of up to 50% of plant-rated electrical load without a reactor trip. The NSSS control systems margin-to-trip analysis confirmed the steam dump system capability at uprated power conditions. There is acceptable margin to the relevant reactor trip setpoints during and following the 50% load rejection transient. To provide effective flow control on large step-load reductions or a plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full-load pressure and steam generator design pressure. The steam dump valves are also required to modulate to control flow. The steam dump valves continue to satisfy these requirements at MUR power uprate conditions.

VI.1.A.iii Extraction Steam System

The extraction steam system heats the condensate and feedwater at various stages prior to the SGs, and provides the normal steam supply to the auxiliary steam system. Based on evaluation results, the extraction steam system operating parameters (pressure, temperature, flow, velocity) are not significantly impacted at MUR power uprate conditions. Therefore, the extraction steam system is acceptable at power uprate conditions.

VI.1.A.iv Condensate and Main Feedwater Systems

The condensate and main feedwater systems are described in UFSAR Section 10.4.3. These systems were evaluated to determine the impact of the MUR power uprate.

VI.1.A.iv.a Condensate System

There are three parallel 50% capacity condensate pumps. Normally two condensate pumps are operating at full load delivering water to the main feedwater pumps suction header. Two low pressure and three high pressure heater drain pumps are normally operating at full load.

The power uprate results in increased condensate flow of approximately 1.9%. Adequate condensate pump net positive suction head is available at uprate conditions. Piping pressures and temperatures are not significantly impacted. Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the condensate system is acceptable at power uprate conditions.

VI.1.A.iv.b Main Feedwater System

There are three parallel motor-driven main feedwater pumps, with two in operation at full load conditions. These pumps are constant speed, so feedwater flow is controlled by the feedwater regulating valves on the pump discharge.

The power uprate results in increased feedwater flow of approximately 1.9%. Adequate main feedwater pump net positive suction head is available at uprate conditions. The increase in extraction steam flow through the feedwater heaters results in a small increase in feedwater temperature entering the SG. Main feedwater isolation valves, feedwater regulating valves, feedwater regulating bypass valves, and main feedwater pump discharge valves provide a containment isolation feature. The existing NSSS accident analysis was completed at 102% of 2893 MWt, which bounds the power uprate. Piping pressures and temperatures are not significantly impacted. Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the main feedwater system is acceptable at power uprate conditions.

VI.1.A.iv.c Abnormal/Transient Operating Conditions

The following transients that impact feedwater flow were evaluated at power uprate conditions: loss of heater drain pump (high pressure or low pressure), loss of a condensate pump, loss of a main feedwater pump, and 50% load rejection.

There is no significant impact on system operation from any of these postulated transients.

VI.1.A.v Feedwater Heaters

There are two parallel trains of feedwater heaters. Each train consists of five heaters (6th, 5th, 4th, 3rd and 2nd point heaters) located on the suction side of the main feedwater pumps. The 6th and 5th point feedwater heaters are located in the main condenser neck. Two additional feedwater heaters (1st points) are located on the discharge side of the main feedwater pumps.

The 6th point heaters tube bundles were replaced on Unit 2 during the Fall 2008 Refueling Outage. Unit 1 replacement is scheduled for the Spring 2009 Refueling Outage. These system changes were independent of the power uprate. The hydraulic calculation evaluated the 6th point heaters based on the replacement tube bundles, because this results in a more conservative pressure drop assessment.

Relevant feedwater heater parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the feedwater heaters are acceptable at power uprate conditions.

VI.1.A.vi Feedwater Heater and Moisture Separator Reheater Vents and Drains

The secondary vent and drain systems are described in UFSAR Section 10.4.6. Feedwater heater and moisture separator reheater vents and drains were evaluated at MUR power uprate conditions. Operating parameters (flow, pressure, temperature, velocity) at power uprate conditions do not significantly impact piping, component, and equipment design parameters. Therefore, feedwater heater and moisture separator reheater vents and drains piping, component, and equipment design are acceptable at MUR power uprate conditions.

VI.1.A.vii Auxiliary Feedwater System

The AFW system design basis of record is described in UFSAR Section 10.4.3. The AFW system serves as a backup system for supplying feedwater to the SGs when the main feedwater system is not available. Each unit's system includes two motor driven pumps and one turbine driven pump configured into three trains. Each pump takes suction through independent lines from the missile protected ECST. The AFW system analyses are based on a core thermal power level of 2951 MWt, which is 102% of 2893 MWt. The analyzed core power level of 2951 MWt remains conservative and bounds the MUR power level. The AFW system maximum operating pressure and temperatures remain essentially unchanged as a result of the MUR power uprate. Piping and component pressure and temperatures design parameters bound power uprate operating pressure and temperature conditions. AFW system flow requirements associated with the analysis are bounding for the power uprate. The AFW system has the capacity to provide adequate flow under transient and accident conditions. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design/performance or operation. Since no changes are being made to the pump design, the brake horse-power requirements are unaffected. No AFW system modifications are required to support the MUR power uprate.

There are two design basis scenarios that define the ECST volume requirements: 8 hours in hot standby (MODE 3) and 2 hours in hot standby (MODE 3) followed by a 4 hour cooldown to RHR entry conditions. The minimum required ECST volume is 93,851 gallons for the design scenario of holding 8 hours in MODE 3 without cooldown. The 8 hour integrated decay heat was based on a core power of 2951 MWt. Therefore, core power remains conservative and bounding for the power uprate. The minimum required ECST volume is 95,988 gallons for the design scenario of 2 hours in MODE 3 followed by a 4 hour cooldown to RHR entry conditions. The current analysis of record uses a core power of 2968.2 MWt, which is 102.6% of 2893 MWt. The power level is bounding for the power uprate. The Technical Specification minimum ECST volume requirement of 110,000 gallons ensures that the usable volume bounds the minimum ECST volume requirement for both scenarios. Therefore, the auxiliary feedwater system is acceptable at power uprate conditions.

VI.1.B Containment Systems

The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the environmental qualification acceptance limits.

VI.1.B.i Containment Quench Spray and Recirculation Spray Systems

The Quench Spray (QS) and Recirculation Spray (RS) systems operate to limit peak containment pressure to less than the design pressure of 45 psig during a LOCA or MSLB, to maintain containment structural integrity. Both systems provide a cooling spray into the containment to remove heat from the containment atmosphere. The QS system takes water from the RWST, mixes in sodium hydroxide from the chemical addition tank to assist in iodine removal and to control containment sump water pH, and delivers the discharge through containment spray rings. The RS system consists of two inside containment and two outside containment subsystems. The RS system takes water from the containment sump and delivers the discharge through containment spray rings.

The existing containment response analyses remain bounding for the power uprate. The QS system and RS system operating and design parameters in the existing analyses bound the power uprate parameters. There are no new operating requirements imposed on either system as a result of the power uprate. Therefore, the QS system and RS system are acceptable for operation at MUR uprate conditions.

VI.1.B.ii Containment Air Cooling

The containment ventilation systems are described in UFSAR Section 9.4.9. The containment ventilation system provides general area cooling and direct cooling to critical components. It also provides the means to purge the containment atmosphere prior to personnel entry during maintenance periods. Containment air cooling consists of a recirculation cooling system, CRDM cooling system, filter system, purge system, and dome air recirculation system. The RCP motor exhaust air is cooled by an integral heat exchanger supplied by the component cooling water system. The CRDM cooling system is discussed in Section VI.1.B.iii. The recirculation and CRDM cooling systems provide air cooling that in combination with the RCP motor cooling maintain containment bulk air temperature within the Technical Specification limits.

CRDM equipment was analyzed at MUR power uprate conditions. There is no heat increase to the containment atmosphere from the CRDM system. NSSS equipment heat load changes were analyzed at MUR power uprate conditions. The heat changes can increase the containment bulk air temperature by about 0.25°F and are considered insignificant. Therefore, the MUR power uprate will have no significant impact on the containment atmosphere.

VI.1.B.iii CRDM Ventilation

The CRDM cooling system was evaluated at power uprate conditions to demonstrate that the electro-magnetic coils design temperature was not exceeded, and to determine the expected additional heat load associated with higher reactor head temperatures.

The lift coil temperature after 15 minutes of stepping is the limiting case for maximum coil temperature. At power uprate conditions, the maximum expected electro-magnetic coil temperature after 15 minutes of stepping is 287.9°F. This is below the coil design temperature of 392°F. The heat load from the CRDMs to the cooling air depends on the fluid temperature underneath the reactor vessel head, T_{head}, increases 0.7°F from the power uprate. This temperature increase is so small that there is no effective increase in the containment heat load.

The CRDM coil operating temperatures remain below their design temperature limits at power uprate conditions, without equipment upgrade or changes in operating parameters. Therefore, the CRDM cooling system is acceptable at MUR power uprate conditions.

VI.1.C Safety-Related Cooling Water Systems

VI.1.C.i Component Cooling Water System

The Component Cooling Water (CCW) system is described in UFSAR Section 9.2.2. The 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 CCW system was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the MUR power uprate heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. The analysis confirms that at MUR uprated conditions, normal plant operation and required cooldown time continue to be met.

VI.1.C.ii Service Water System

The Service Water (SW) system is described in UFSAR Section 9.2.1 and is common to both units. There are 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 power uprate heat removal requirements during normal, shutdown, and accident conditions. The evaluations determined that the existing SW flows will continue to support the heat removal requirements at uprate conditions. The SW system and component design parameters remain bounding for power uprate operation. No system modifications are required to support the power uprate. Therefore, the SW system is acceptable for operation at power uprate conditions.

VI.1.C.iii Ultimate Heat Sink

The ultimate heat sink is described in UFSAR Section 9.2.5 and is common to both units. The Technical Specification required ultimate heat sink is the SW Reservoir.

The SW system inlet temperature for normal, shutdown, and accident conditions is bounded for the power uprate. 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 power uprate. Therefore, the ultimate heat sink is acceptable for operation at power uprate conditions.

VI.1.C.iv Residual Heat Removal System

NAPS UFSAR Section 5.5.4 describes the Residual Heat Removal (RHR) system. RHR 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 analysis showed that each of these cases met the cooldown time requirements.

VI.1.D Spent Fuel Pool Storage and Cooling Water

NAPS UFSAR Section 4.3.2.7 describes the SFP criticality analysis, with additional information specific to AMBW fuel provided in Section 4.5.3.2.7. NAPS UFSAR Section 9.1.3 describes the SFP cooling and purification system. This system is common to both North Anna units.

VI.1.D.i Spent Fuel Pool Criticality

The analysis of record was submitted to the NRC in Reference VI-1, with additional information provided in References VI-2 through VI-4. The NRC approved the analysis and associated Technical Specification changes in Reference VI-5.

Dominion performed an evaluation to determine the MUR power uprate impact on the SFP criticality analysis of record. The power uprate has no effect on the fresh fuel characteristics, so this portion of the analysis is unaffected. For irradiated fuel, the key effects are higher fuel burnup (due to higher power operation) and higher soluble boron concentration (due to slightly higher energy fuel loadings). These effects result in a SFP k_{eff} increase over the existing analysis. However, this k_{eff} increase is accommodated by crediting other allowed compensating input changes within the methodology constraints and limitations. The primary compensating input change is crediting discrete burnable poison rods use for only one cycle in fresh fuel, rather than assuming the two consecutive cycles of use in the original analysis. There is no operational impact, because discrete burnable poison rods are not used for more than one cycle in current core designs. With this provision, the analysis of record remains bounding and the existing Technical Specification limitations on SFP maximum fuel enrichment and minimum fuel burnup remain acceptable for fuel operated under power uprate conditions.

VI.1.D.ii Spent Fuel Pool Cooling and Purification

SFP cooling heat exchangers are cooled by component cooling water, with service water available as an emergency backup. Heat exchanger outlet flow is sent to the refueling purification system consisting of two filters and an ion exchanger, and then returns to the SFP.

There are no changes to the SFP cooling system limiting temperatures, pressures or flow rates as a result of the power uprate. Uprate conditions are bounded by the existing system design conditions. System modifications are not required to support the power uprate. 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 power uprate conditions.

VI.1.E Radioactive Waste Systems

VI.1.E.i Gaseous Waste

The gaseous waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive gases released during simultaneous operation of both units. Gaseous waste system functions and the volume of waste gas processed are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The gaseous waste system is bounded by the existing system design parameters and is acceptable at power uprate conditions.

VI.1.E.ii Liquid Waste

The liquid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive liquids produced during simultaneous operation of both units. Liquid waste system functions and the liquid waste processed volume are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The liquid waste system is bounded by the existing system design parameters and is acceptable at power uprate conditions.

VI.1.E.iii Solid Waste

The solid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive solid waste produced during simultaneous operation of both units. Solid waste system functions and the solid waste processed volume are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The solid waste system is bounded by the existing system design parameters and is acceptable at power uprate conditions.

VI.1.E.iv Steam Generator Blowdown

The required SG blowdown flow rates during plant operation are based on chemistry control and tubesheet sweep necessary to control solids buildup. The SG blowdown system was analyzed for a blowdown flowrate increase of approximately 4 gpm. However, NAPS will continue to operate the SG blowdown system per the plant chemistry program following the power uprate, with no change in blowdown flowrate. Blowdown system operating temperatures and pressures will decrease and remain bounded by the existing design parameters under uprate conditions.

The uprate will not significantly increase the potential for FAC on the blowdown system piping and components. NAPS will continue to monitor the blowdown system for FAC. Therefore, the SG blowdown system will continue to meet system design requirements at MUR power uprate conditions.

VI.1.F Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning

VI.1.F.i Control Room Ventilation System

NAPS UFSAR Section 9.4.1 describes the main control room and relay rooms heating, cooling and ventilation systems, including main control room and emergency switchgear room chilled water. The main control room and emergency switchgear rooms envelope has two independent air conditioning systems consisting of two air handling units (one for the main control room and one for the emergency switchgear room), chilled water piping and a water chiller (one chiller for one train and two chillers for the other train). The main control room/emergency switchgear room chilled water systems are independent of the station chilled water system. The main control and computer room air conditioning is designed to maintain 75°F and approximately 50% relative humidity in the associated rooms during normal conditions. The relay rooms are designed for 75°F dry bulb at approximately 50% relative humidity in the associated rooms during normal operation. During emergency conditions, the main control room/emergency switchgear room are maintained below the design maximum temperature of 120°F.

The heat loads (electrical heat loads, lighting, personnel) at MUR power uprate conditions were evaluated. Radiological consequences of the MUR are discussed in Section II.2.29. 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 power uprate conditions, because the heat loads in these areas do not increase.

VI.1.F.ii ESF Ventilation System

NAPS UFSAR Section 9.4.6 describes the ESF areas heating, cooling and ventilation systems. A separate ESF ventilation subsystem is provided for NAPS Units 1 and 2. The ESF areas ventilation subsystems are designed to limit temperatures to 120°F during warm weather and to raise incoming outside air to a minimum temperature of 75°F during cold weather.

The current limiting case heat loads have been evaluated at the MUR power uprate conditions. The safeguards area, quench spray pump house, and rod drive room subsystems ventilation capabilities are not impacted, because there is no increase in electrical heat loads and no significant increase in piping system heat loads at the MUR power uprate conditions. There is a small heat load increase from the feedwater piping and small heat load decrease from the main steam piping resulting in no significant impact in main steam valve house ambient air temperature.

VI.1.F.iii Fuel Handling Area Ventilation System

NAPS UFSAR Section 9.4.5 describes the fuel building ventilation system. The fuel building ventilation system is a once through ventilation system that provides the fuel building with 100% outdoor air, after it has been filtered and heated as required. The ventilation system consists of two supply fans, one that serves the SFP area and one for the remote equipment space at elevation 249 feet 4 inches. Both supply fans take suction from a common plenum fitted with a combination roll and high efficiency filters and steam coils for air tempering and space heating. The ventilation system maintains a maximum air temperature of 105°F and a minimum air temperature of 75°F.

The SFP cooling equipment loads analyses are not impacted by the MUR power uprate. As discussed in Section II.2.37, the higher decay heat loads will not impact the limiting case full core off-load. The maximum SFP and piping temperatures at MUR conditions will be at or below the calculated limiting case. The fuel building ventilation system is not impacted by the MUR power uprate, because there is no increase in the SFP temperature, piping or electrical heat loads.

VI REFERENCES

VI-1 Letter from Leslie N. Hartz (Dominion) to USNRC Document Control Desk, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes, Increased Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit,"* Serial No. 00-491, ML003758403, September 27, 2000.

VI-2 Letter from William R. Matthews (Dominion) to USNRC Document Control Desk, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes, Increased Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit, Corrected Tables*, Serial No. 01-051, ML010430321, February 2, 2001.

VI-3 Letter from William R. Matthews (Dominion) to USNRC Document Control Desk, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes, Increased Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit, Request for Additional Information*, ML010680301, Serial No. 01-051A, March 2, 2001.

VI-4 Letter from Leslie N. Hartz (Dominion) to USNRC Document Control Desk, *Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Implementation of Proposed Technical Specifications Changes, Increased Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit*, Serial No. 01-294, ML011450173, May 21, 2001.

VI-5 Letter from Stephen Monarque (USNRC) to David A. Christian (Dominion), *Issuance of Amendments Re: Technical Specifications Changes to Increase Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit (TAC Nos. MB0197 and MB0198)*, ML011700557, June 15, 2001.

VII. OTHER

1. A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.

2. A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

   A. emergency and abnormal operating procedures.

   B. control room controls, displays (including the safety parameter display system) and alarms.

   C. the control room plant reference simulator.

   D. the operator training program.

3. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.

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

5. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

   A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

   B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

RESPONSE TO VII - OTHER

VII.1 Operator Actions

Operator actions included in the safety analyses were reviewed for potential MUR power uprate impact. The following design basis events were reviewed:

Appendix R Fire	UFSAR Section 9.5.1[(1)]
Boron Dilution	UFSAR Section 15.2.4
Startup of Inactive Reactor Coolant Loop	UFSAR Section 15.2.6
Small Break LOCA	UFSAR Section 15.3.1
VCT Rupture	UFSAR Section 15.3.6
Large Break LOCA	UFSAR Section 15.4.1
Main Steamline Break	UFSAR Section 15.4.2.1
Main Feedwater Line Break	UFSAR Section 15.4.2.2
Steam Generator Tube Rupture	UFSAR Section 15.4.3
Fuel Handling Accident	UFSAR Section 15.4.5

1. The Appendix R safe shutdown analyses are not described in UFSAR Section 9.5.1, where fire protection equipment is specified. The post-fire safe shutdown analyses are maintained in engineering calculations

The safety analysis reviews have determined that the existing required operator actions are not affected by the MUR power uprate. There is no reduction in time for required operator actions. No new manual operator actions were created and no existing manual actions were automated.

The power uprate is being implemented under the administrative controls of the design change process. Other potential impacts on operator actions and action times in plant procedures may be identified and evaluated during the design change impacts review. The design change process ensures that impacted procedures will be revised prior to the power uprate implementation.

VII.2.A Emergency and Abnormal Operating Procedures

Emergency and abnormal operating procedures were reviewed to determine any MUR power uprate impact. No changes are required to the procedure steps and mitigation actions as a result of the MUR power uprate. However, the review identified three emergency operating procedure (EOP) setpoints that require revision, because these setpoints were developed at 2893 MWt. Using core RTP to develop these setpoints is consistent with the Westinghouse Owners Group background document.

These EOP setpoints will be revised to reflect a total core power of 2951 MWt, which is 102% of 2893 MWt and bounds the MUR power uprate. The procedure changes and any associated operator training will be completed during power uprate implementation and prior to operation above 2893 MWt.

There are no operator action changes for shutdown risk management due to MUR power uprate. The time to core boil will decrease due to the MUR but the method of calculating the time to core boil will remain the same. NAPS procedures will be revised with data generated with decay heats at the MUR power level. Operator training on the procedure changes will be provided as part of the MUR implementation.

VII.2.B Control Room Controls, Displays and Alarms

The following changes/modifications associated with the proposed power uprate affect control room controls.

- Instruments associated with turbine first stage pressure will require scaling changes for NSSS protection permissive P-13 and control permissives C-5, C-7 and C-20.

The following modifications associated with the proposed power uprate affect operator displays (including the safety parameter display system (SPDS)).

- Instrument loops are affected by the power uprate (indicator replacement, calibration span, and/or scaling).
- Plant computer points will be added and/or changed for the revised calorimetric algorithm and the feedwater ultrasonic flow meter.
- No significant SPDS changes are anticipated as a result of the MUR power uprate. Critical safety function status trees will be reviewed and revised as necessary.
- The new 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 operations.

The following modifications associated with the proposed power uprate affect alarms.

- 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.

VII.2.C Control Room Plant Reference Simulator

The MUR power uprate is being implemented under the plant modification process administrative controls. As part of this process, potential simulator modifications will be identified. Simulator required changes resulting from the MUR power uprate will be evaluated, implemented and tested per NAPS approved procedures. Simulator fidelity will be revalidated per NAPS approved procedures. Any required simulator modifications will be completed in time to support operator training prior to MUR power uprate implementation.

VII.2.D Operator Training Program

The operator training program requires revision as a result of the MUR power uprate. Operator training will be developed and the operations staff will be trained on the plant modifications, Technical Specification and TRM changes, and procedure changes prior to MUR power uprate implementation.

VII.3 Intent To Complete Modifications

Dominion will complete the modifications required to support the MUR (including operator training) prior to power uprate implementation.

VII.4 Temporary Operation Above Licensed Power Level

Dominion will revise the existing plant operating guideline related to temporary operation above full steady-state licensed power levels. Precautions will be revised to account for the uprate power level.

VII.5 10 CFR 51.22 Discussion

VII.5.A 10 CFR 51.22 provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (A.1) involve a significant hazards consideration, (A.2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (B) result in a significant increase in individual or cumulative occupational radiation exposure.

Dominion has determined that this license amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment is required in connection with issuance of the proposed license amendment. The basis for this determination follows:

1. The proposed license amendment does not involve a significant hazards consideration as previously described in Attachment 1, Section 5.0 Regulatory Analysis for this License Amendment Request.

2. The proposed change does not involve installing new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite.

   There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level.

   Gaseous and liquid radwaste effluent activity is expected to increase from current levels by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and 10 CFR 50, Appendix I limits per the North Anna Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

   Solid radwaste effluent activity is expected to increase from current levels proportionately to the increase in long half-life coolant activity. The total long-lived activity is bounded by the percent of power uprate. Changes in solid waste volume are not expected.

   Therefore, the license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

VII.5.B The license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage of core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within acceptable limits by the site Radiation

Protection Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

UFSAR Section 9.5.1 describes the NAPS Fire Protection Program. The Fire Protection Program satisfies the regulatory criterion of General Design Criteria 3; 10 CFR 50, Appendix R (Sections III.G, III.J, III.L, and III.O); and Branch Technical Position APCSB 9.5-1, Appendix A.

VII.6.A.i Fire Protection Systems

The Fire Protection System consists of the following major subsystems: fire detection (including smoke detectors, heat detectors, alarms), water suppression (including fire pumps, main fire loop piping, sprinkler systems, deluge systems), CO_2 suppression, Halon suppression, manual fire equipment (portable fire extinguishing equipment), and fire barriers (including fire walls, fire doors, penetration seals, cable wraps, cable tray stops, heat shields). The fire protection subsystems remain unchanged as a result of the MUR power uprate.

VII.6.A.ii Responsibilities

Plant management, supervisory and station personnel responsibilities in support of the Fire Protection Program are not impacted by the MUR power uprate.

VII.6.A.iii Administrative Controls

Topics include control and use of fire protection systems and equipment; combustibles storage; control of ignition sources; implementing ventilation for heat and smoke removal; design change control for fire protection systems and equipment; Fire Protection Program instructions, procedures, and drawings; fire inspection program; fire equipment maintenance and testing; and fire strategies. The MUR power uprate does not affect the established administrative controls.

VII.6.A.iv Fire Brigade

There are no changes in the fire brigade structure, responsibilities, reporting relationships, or qualifications resulting from the MUR power uprate.

VII.6.A.v Evaluations of Inadvertent Operation of Fire Protection Systems

The MUR power uprate does not affect the existing evaluation conclusions for the inadvertent operation of fire protection systems.

VII.6.B High Energy Line Break Program

The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. UFSAR Appendix 3C, "Effects of Piping System Breaks Outside Containment," describes the high and moderate energy line break analysis. High-energy pipe breaks are analyzed for piping for which the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200°F. High-energy pipe cracks are postulated in piping for which either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200°F.

The evaluation concluded that the MUR power uprate does not result in any new or revised high or moderate energy line break locations. The high and moderate energy line break analysis is not affected. Area temperature and pressure resulting from high energy line breaks and internal flooding conditions resulting from moderate energy line breaks remain valid at power uprate conditions.

VII.6.C Appendix J Program

UFSAR Section 6.2.1.4.1 Containment Leakage Tests, states that a performance based testing program will include Type A tests to measure the containment overall integrated leakage rate, Type B tests to detect and measure local leakage from certain containment components, and Type C tests to measure containment isolation valve leakage rates. The containment leakage tests are performed as required by 10 CFR 50, Appendix J, Option B.

A review of the LOCA response analysis confirmed that the analysis was performed at 102% of 2893 MWt. Because the LOCA peak pressure analysis is unaffected, P_a at MUR power uprate conditions is unchanged from the current conditions specified in NAPS Technical Specification 5.5.15. No changes or modifications are required to the existing Appendix J Program or procedures. Therefore, NAPS Technical Specification 5.5.15 and the applicable NAPS Appendix J Program procedures are acceptable at MUR uprate conditions.

VII.6.D Coatings Program

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and radionuclide contamination. Coatings also provide wear protection during plant operation and maintenance activities. These coatings are subject to 10 CFR 50, Appendix B quality assurance requirements, because their degradation could adversely impact safety related equipment. The approved NAPS containment Service Level 1 coatings are qualified to withstand a LOCA environment and meet ANSI Standards N5.12, N101.2 and N101.4.

The UFSAR LOCA containment response analyses remain bounding for the MUR power uprate. There were no changes to the containment analyses that would require a change to the containment design pressure or temperature. Since the containment design pressure and temperature limits were used to qualify the Service Level 1 containment coatings, and those limits are not changing, the Service Level 1 containment coatings remain qualified under MUR power uprate conditions.

VII.6.E NRC Generic Letters

The design criteria for safety-related valves are promulgated in 10 CFR 50.55a. Additional information is also provided by the plant specific evaluations of GL 89-10, GL 95-07 and GL 96-06. The plant specific provisions of GL 89-10, GL 95-07, and GL 96-06 were reviewed to determine if any changes were required as a result of the power uprate. No required changes were identified.

VII.6.E.i GL 89-10 Motor Operated Valve (MOV) Program

The NRC issued GL 89-10 (Reference VII-1) requiring licensees to develop a comprehensive program to ensure MOVs in safety-related systems would operate under design basis conditions.

The review determined that the maximum differential pressures/line pressures determined in the system and functional design basis review calculations for the GL 89-10 identified MOVs were not affected by the MUR power uprate. The values for these parameters at current conditions bound the values at MUR conditions. Therefore, these parameters do not affect the calculations that determine MOV thrust and torque values. The MOV flow rates documented in the system and functional design basis review calculations for the GL identified MOVs at current conditions bound the flow rates at MUR conditions. The MUR does not affect the maximum ambient temperatures used to determine MOV motor capability torque values at current conditions. The MUR power uprate has no effect on valve factors or required thrusts because pressure, temperature and flow conditions are not a direct input into calculating the valve factor. Therefore the conclusions previously provided for GL 89-10 identified MOVs are not impacted by the MUR power uprate.

VII.6.E.ii GL 95-07 Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves

The NRC issued GL 95-07 (Reference VII-2) to address potential pressure locking and thermal binding of safety-related power operated gate valves. NAPS responded to this GL in References VII-3 and VII-4.

The review determined that the MUR power uprate does not affect the pressure locking evaluations previously completed. The thrust required to open the applicable valves remains less than the motor actuator capabilities at MUR conditions. The power uprate does not affect valve design, valve function, or operational conditions. New conditions were not created that would affect valve susceptibility to pressure locking or thermal binding. Therefore, the conclusions previously provided in Reference VII-4 for valve pressure locking and thermal binding acceptability are not impacted by the MUR power uprate.

VII.6.E.iii GL 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The NRC issued GL 96-06 (Reference VII-5) to address hydrodynamic effects of waterhammer and two-phase flow conditions on cooling systems serving containment air coolers and thermally induced overpressurization of isolated piping segments. NAPS responded to this GL in References VII-6 and VII-7.

Containment air cooling system two-phase flow and water hammer are not applicable, because the system is isolated and de-energized during design basis accidents. The MUR power uprate does not modify system configuration or change system operation. The QS and RS systems piping is not filled with water until after a containment depressurization actuation signal. Thus, system piping overpressurization cannot occur prior to system actuation. The QS and RS systems are not modified and system operating parameters are unchanged. The current LOCA accident analyses were performed at 102% of 2893 MWt and remain bounding for the MUR power uprate. There is no increase in the possibility of overpressurizing isolated segments of safety-related piping inside containment, including penetrations, as a result of the power uprate. Therefore, there is no impact regarding GL 96-06 program issues at power uprate conditions.

VII.6.F Air Operated Valve Program

The NAPS air operated valve (AOV) Program includes the following categories of AOVs:

Category 1 – AOVs that are high safety significant

Category 2 – AOVs that are low safety significant, safety-related, and non-safety qualified

The system evaluations for Category 1 AOVs indicate that the MUR does not affect the maximum differential pressures/line pressures, flow rates, or fluid temperatures documented in the system level design basis review calculations. Therefore, the MUR power uprate does not affect the AOV setup values in the component level calculations for these AOVs.

The system evaluations for Category 2 AOVs indicate that the existing maximum operating flow rates and pressures are bounded by the current parameters and remain valid at MUR power uprate conditions.

VII REFERENCES

VII-1 NRC Generic Letter 89-10, *Safety-Related Motor Operated Valve Testing and Surveillance*, June 28, 1989 and supplements.

VII-2 NRC Generic Letter 95-07, *Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves*, August 17, 1995.

VII-3 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2 Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves*, Serial No. 95-566, November 15, 1995.

VII-4 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2 Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves*, Serial No. 95-566A, February 7, 1996.

VII-5 NRC Generic Letter 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions*, September 30, 1996.

VII-6 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2 NRC Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions*, Serial No. 96-516, October 30, 1996.

VII-7 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2 NRC Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity*, Serial No. 96-516A, January 28, 1997.

VIII. CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

   A. a description of the change

   B. identification of analyses affected by and/or supporting the change

   C. justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change

RESPONSE TO VIII - CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

VIII.1 Technical Specification Changes

VIII.1.A Description of Change

Table VIII-1
Description of Technical Specifications Changes

Change No.	Change Description
1	**Facility Operating License, Paragraph 2.C(1)** Dominion is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal)
2	**TS Section 1.1, Definitions – Rated Thermal Power** RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2940 MWt.

The following information provides the supporting justification for the proposed Technical Specifications changes described above.

VIII.1.B Supporting Analysis

The current NAPS RTP is 2893 MWt. The MUR power uprate will increase power by approximately 1.6%. This increase is based on a plant specific evaluation of reactor power measurement uncertainty using the UFM instrumentation versus the 10 CFR 50, Appendix K previously mandated 2 percent uncertainty. Therefore, the new RTP will be:

RTP = 2893 MWt * 1.01625 = 2940 MWt (rounded down for conservatism)

VIII.1.C Justification for Changes

Detailed evaluations and analyses were performed demonstrating that NAPS operation at a reactor power level of 2940 MWt is acceptable. The detailed evaluations and analyses considered the effects of operation at this power level on: power level measurement uncertainty; postulated accidents and transients; mechanical, structural and material components integrity and design; electrical equipment design; system design; operator actions, emergency and abnormal operating procedures, control room, plant simulator, and operator training; environmental impact; and Technical Specifications, protection system settings, and emergency system settings.

The evaluations and analyses were performed using current licensing basis acceptance criteria and Technical Specifications. This ensures the same protection level for public health and safety at the uprated conditions as the currently licensed power level. These evaluations and analyses are described in this attachment. Attachment 2 contains the Operating License and Technical Specification marked-up pages and Attachment 3 contains the typed pages to reflect the proposed changes.

VIII.2 Protection System Settings Changes

There are no protection system setpoint changes resulting from this LAR, although some instruments will require rescaling to support MUR implementation.

VIII.3 Emergency System Settings Changes

There are no emergency system setpoint changes resulting from this LAR, although some instruments will require rescaling to support MUR implementation.

ATTACHMENT 6

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

LIST OF REGULATORY COMMITMENTS

NORTH ANNA POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

REGULATORY COMMITMENTS

The following list identifies those actions committed to by NAPS in this LAR. Any other actions discussed in the submittal represent intended or planned actions described for information only and are not regulatory commitments.

COMMITMENT	SCHEDULED COMPLETION DATE (if required)
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.
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).

COMMITMENT	SCHEDULED COMPLETION DATE (if required)
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).