



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

July 27, 2018

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS REGARDING NEW AND SPENT FUEL STORAGE (CAC NOS. MF9712 AND MF9713; EPID L-2017-LLA-0240)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 279 and 262 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS) Unit Nos. 1 and 2, respectively. These amendments are in response to your application dated May 2, 2017, as supplemented by letters dated July 19, 2017, and January 31, 2018. The amendments modify the NAPS Technical Specifications 3.7.18, "Spent Fuel Pool Storage," and TS 4.3.1, "Criticality," to allow the storage of fuel assemblies with a maximum enrichment of up to 5.0 weight percent uranium 235 in the NAPS spent fuel pool storage racks and the New Fuel Storage Area. The amendments further revise the allowable fuel assembly parameters and fuel storage patterns in the spent fuel pool.

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

Sincerely,

A handwritten signature in cursive script that reads "James R. Hall".

James R. Hall, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:

1. Amendment No. 279 to NPF-4
2. Amendment No. 262 to NPF-7
3. Safety Evaluation

cc: via Listserv



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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 279
Renewed License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated May 2, 2017, as supplemented by letters dated July 19, 2017, and January 31, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to paragraph 2.C (2) of Renewed Facility Operating License No. NPF-4, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shaun Williams for

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Operation

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

Date of Issuance: July 27, 2018



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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 262
Renewed License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated May 2, 2017, as supplemented by letters dated July 19, 2017, and January 31, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to paragraph 2.C (2) of Renewed Facility Operating License No. NPF-7, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 262, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shawn Williams for

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Operation

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

Date of Issuance: July 27, 2018

ATTACHMENT TO
NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2
LICENSE AMENDMENT NO. 279
RENEWED FACILITY OPERATING LICENSE NO. NPF-4
DOCKET NO. 50-338
AND LICENSE AMENDMENT NO. 262
RENEWED FACILITY OPERATING LICENSE NO. NPF-7
DOCKET NO. 50-339

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove

NPF-4, page 3
NPF-7, page 3

Insert

NPF-4, page 3
NPF-7, page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove

3.7.18-1
3.7.18-2
3.7.18-3
3.7.18-4

4.0-1
4.0-2

Insert

3.7.18-1
3.7.18-2
3.7.18-3
3.7.18-4
3.7.18-5
4.0-1
4.0-2

- (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

Technical Specifications contained in Appendix A, as revised through Amendment No. 279 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 component; 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. 262 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 insurance of the condition or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

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

3.7 PLANT SYSTEMS

3.7.18 Spent Fuel Pool Storage

LCO 3.7.18 The combination of initial enrichment, burnup and cooling time of each fuel assembly stored in the fuel storage pool shall be in accordance with the following:

a. Region 1 Fuel Storage Locations:

- I. Fuel assemblies stored in Region 1 shall be stored in a 2-out-of-4 checkerboard arrangement with empty cells per Figure 3.7.18-1:
 - i. Empty cells shall remain empty with the exception of a Rod Cluster Control Assembly (RCCA) and/or a cell blocker.
- II. A Region 1 checkerboard is a rectangle of assemblies that can be placed anywhere in the spent fuel pool with the following restrictions:
 - i. All 4 corners of a Region 1 block shall be an empty cell location.
 - ii. There shall be a minimum of two (2) Region 2 rows between two Region 1 blocks.
 - iii. Region 1 shall NOT cross a spent fuel rack module boundary.
 - iv. Spent Fuel Pool Locations AA21, AA22, BB21, BB22, CC21, and CC22 shall NOT be contained in a Region 1 block.
- III. There are no restrictions on burnup and cooling time on fuel of initial enrichment of less than or equal to 5.0 weight percent (wt%) U-235

b. Region 2 Fuel Storage Locations:

- I. Irradiated fuel assemblies with a combination of initial enrichment and burnup in the "Acceptable" burnup domain in Figure 3.7.18-2 may be stored in Region 2.
- II. Irradiated fuel assemblies cooled three (3) or more years with a combination of initial enrichment and burnup in the "Acceptable" burnup domain in Figure 3.7.18-3 may be stored in Region 2.

(continued)

1. Rack modules that are adjacent to the spent fuel pool wall may credit the wall region as empty cells for the purposes of meeting the Region 1 requirements of LCO 3.7.18.a.II.i and 3.7.18.a.II.iii.

b. (continued)

----- NOTE -----

Regarding fuel assemblies that contain a full length RCCA - if the enrichment, burnup, and cooling time of such an assembly stored in Region 2 is NOT in the "Acceptable" burnup domain in Figure 3.7.18-2 or 3.7.18-3 (e.g., the assembly requires a full length RCCA for storage in Region 2), then the assembly must be in a Region 1 storage location when its RCCA is inserted or removed.

III. There are no restrictions on initial enrichment, burnup, and cooling time on a fuel assembly stored in Region 2 if the assembly contains a full length RCCA.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	<p>A.1 -----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>Initiate action to move the noncomplying fuel assembly to an acceptable location.</p>	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.18.1 Verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, cooling time, RCCA placement, and location of the assembly are acceptable.	Prior to storing the fuel assembly in the spent fuel pool

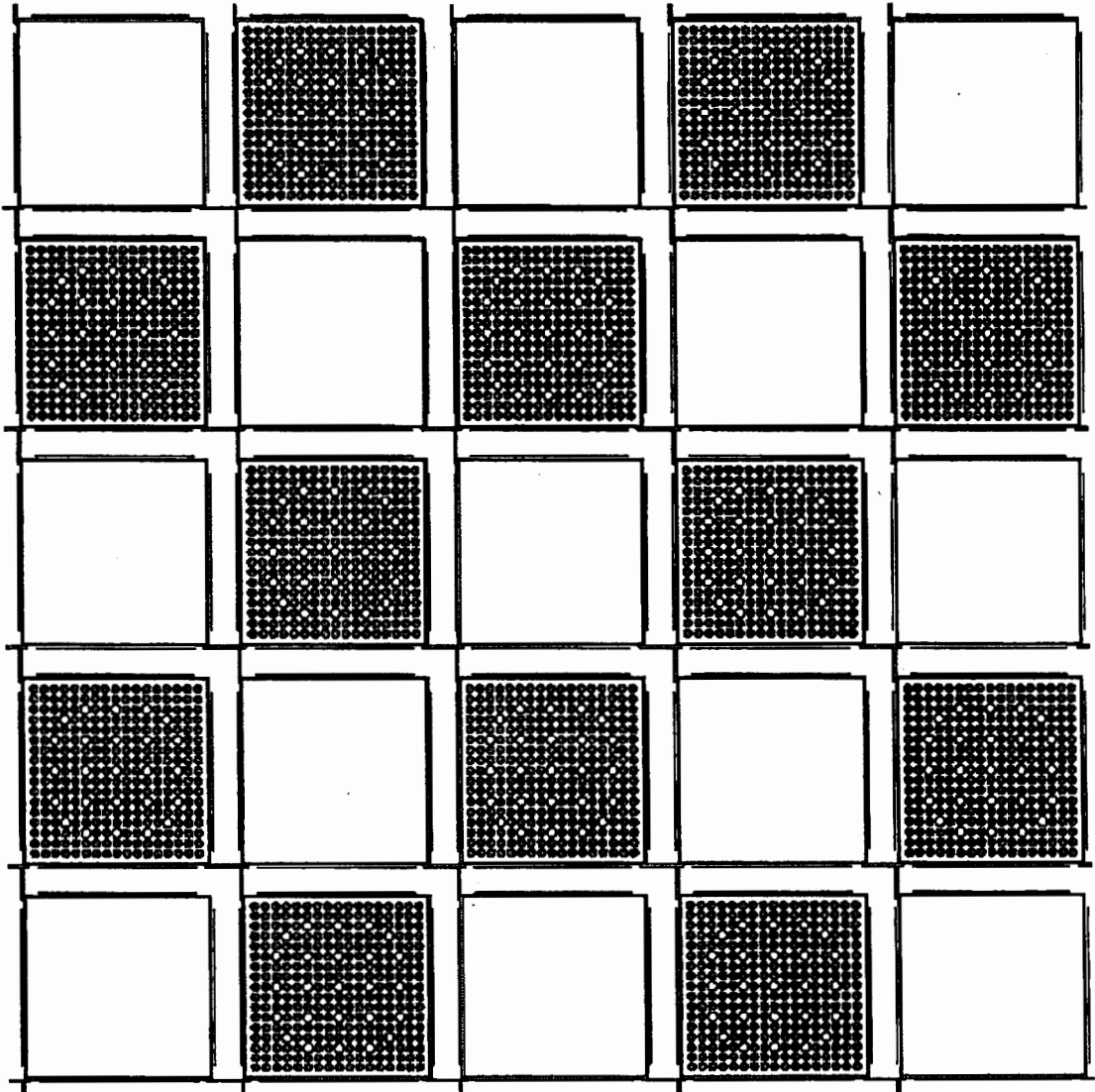


Figure 3.7.18-1 (page 1 of 1)
Typical Region 1 Checkerboard

Spent Fuel Pool Storage
3.7.18

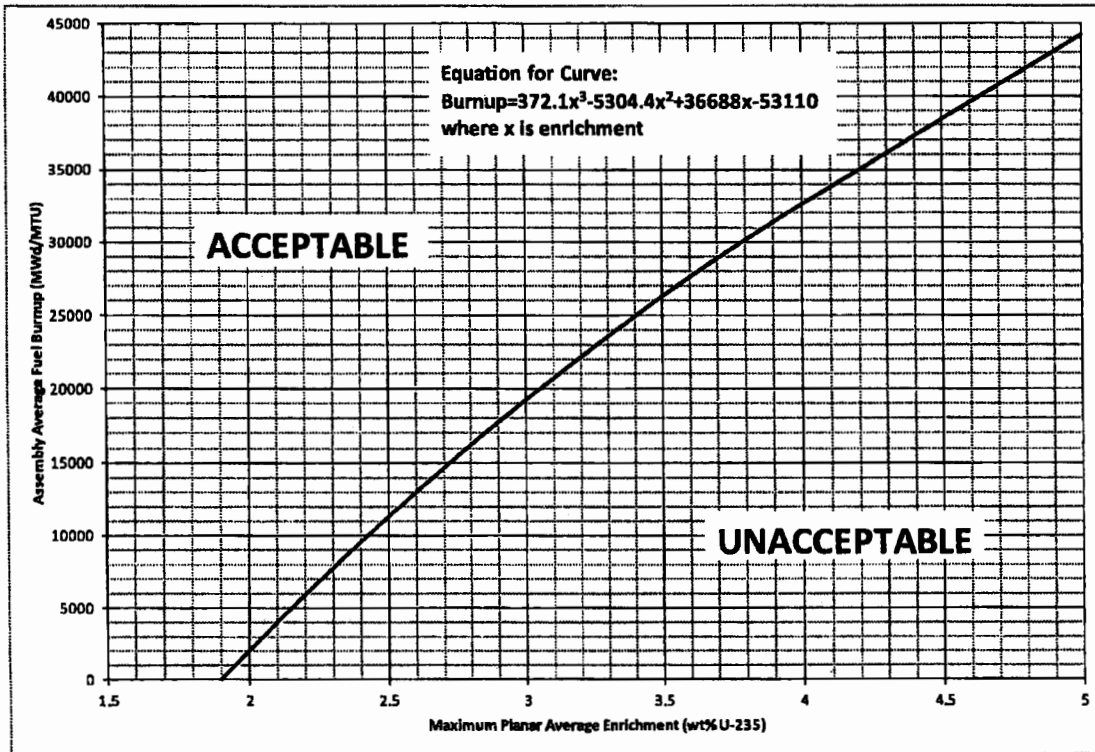


Figure 3.7.18-2 (page 1 of 1)
Minimum Burnup Requirements for Region 2 With No Credit for Cooling

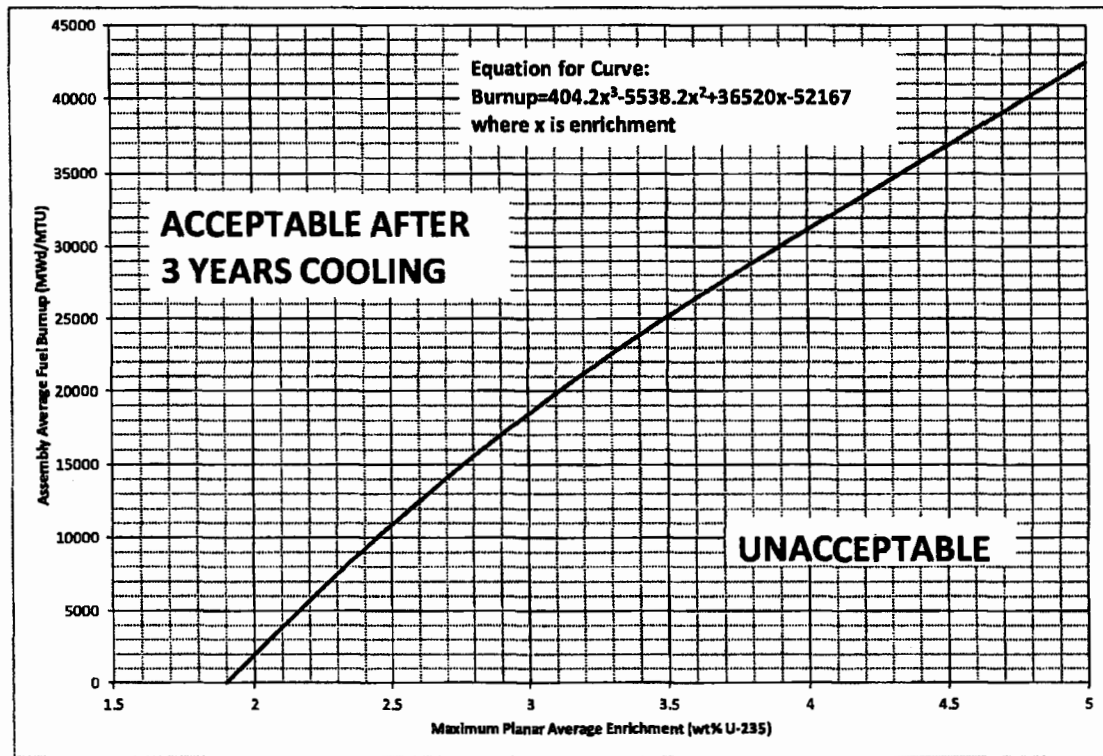


Figure 3.7.18-3 (page 1 of 1)
Minimum Burnup Requirements for Region 2 for Assemblies Cooled
3 Years or More

4.0 DESIGN FEATURES

4.1 Site Location

The North Anna Power Station is located in the north-central portion of Virginia in Louisa County and is approximately 40 miles north-northwest of Richmond, 36 miles east of Charlottesville; 22 miles southwest of Fredericksburg; and 70 miles southwest of Washington, D.C. The site is on a peninsula on the southern shore of Lake Anna at the end of State Route 700.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO, Optimized ZIRLO, or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 48 control rod assemblies. The control material shall be silver indium cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

4.0 DESIGN FEATURES

4.3.1.1 (continued)

- b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties and biases calculated in accordance with the methodology described in UFSAR Section 9.1;
- c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 900 ppm, which includes an allowance for uncertainties and biases calculated in accordance with the methodology described in UFSAR Section 9.1; and
- d. A nominal 10 9/16 inch center to center distance between fuel assemblies placed in the fuel storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties and biases;
- c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties and biases; and
- d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 285 feet, 9 inches, Mean Sea Level, USGS datum.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1737 fuel assemblies.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 279 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 262 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated May 2, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17129A446), as supplemented by letters dated July 19, 2017, and January 31, 2018 (ADAMS Accession Nos. ML17207A161 and ML18037A704, respectively), Virginia Electric and Power Company (Dominion, or the licensee) submitted a license amendment request (LAR) for the North Anna Power Station (NAPS), Unit Nos. 1 and 2. The proposed license amendments revise the NAPS Unit Nos. 1 and 2 Technical Specifications (TSs) 3.7.18, "Spent Fuel Pool Storage" and TS 4.3.1 "Criticality," to allow the storage of fuel assemblies with a maximum enrichment of up to 5.0 weight percent (w/%) uranium 235 (U-235) in the spent fuel pool (SFP) storage racks and the New Fuel Storage Area (NFSA). The proposed amendments would further revise the allowable fuel assembly parameters and storage patterns for fuel in the SFP.

The supplemental letters dated July 19, 2017, and January 31, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 6, 2018 (83 FR 9553).

2.0 REGULATORY EVALUATION

In accordance with the licensee's amendment request, the regulatory requirements and guidance which the NRC staff considered in assessing the proposed TS change are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, Criterion 62, "Prevention of criticality in fuel storage and handling," requires that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Paragraph 10 CFR 50.68(a), "Criticality accident requirements," states that each holder of an operating license shall comply with either 10 CFR 70.24 or the requirements in 10 CFR 50.68(b). The licensee has elected to meet 10 CFR 50.68(b). Accordingly, and as relevant to this license amendment request, the licensee must comply with the following 50.68(b) requirements:

- (1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.
- (2) The estimated ratio of neutron production to neutron absorption and leakage (k -effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.
- (3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k -effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.
- (4) If no credit for soluble boron is taken, the k -effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k -effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k -effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(4), the TSs will include design features which are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of 10 CFR 50.36.

The NRC staff also considered the following staff guidance documents in its review of the amendment request:

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "Spent Fuel Storage."

NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998 (ADAMS Accession No. ML003728001).

Interim Staff Guidance (ISG) document DSS-ISG-2010-01, "Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01, Rev. 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," dated September 2011 (ADAMS Accession No. ML110620086).

3.0 TECHNICAL EVALUATION

3.1 Background

The key element of the amendment request is the proposal to increase the maximum licensed fuel assembly enrichment from 4.6 w/% U-235 to 5.0 w/% U-235. The changes to the SFP and NFSA storage requirements are necessary to ensure NAPS continues to meet the regulatory requirements for storage of new and spent nuclear fuel with the increased U-235 enrichment.

The NFSA is a concrete structure within the NAPS Auxiliary Building. The NFSA is not watertight, therefore, flooding is not precluded. The racks in the NFSA are stainless steel tubes mounted to a stainless steel framework of L-beams and I-beams. The stainless steel tubes are mounted with a 53.340 centimeter pitch. The construction allows for drainage of the stainless steel tubes. The NFSA capacity is 126 fresh fuel assemblies.

The SFP contains 16 storage arrays of different sizes with a total storage capacity of 1737 fuel assemblies. Each array consists of a welded assembly of individual storage cells. The storage cells are comprised of double-wall Type 304 stainless steel boxes welded to each other with tie plates to maintain the cell pitch of 10-9/16 inches. Each storage cell has an interior height of 168 inches to ensure that the top nozzle and core components do not extend above the top of the spent fuel rack when the fuel assembly is fully inserted. When originally installed, the SFP storage arrays contained the neutron absorbing material Boraflex. Boraflex is known to degrade in SFP environments. In a previous license amendment, the licensee stopped taking credit for the Boraflex, but did not physically remove the Boraflex. This proposed license amendment does not take credit for any remaining Boraflex. The licensee has stated that the residual Boraflex adds margin. There is likely residual Boraflex in the NAPS SFP. With the Boraflex degradation mechanisms there are likely Boraflex panels with little degradation, but there are also likely Boraflex panels that have degraded to the point of uselessness. Since the licensee does not have a Boraflex monitoring program, it does not know the condition, content, or even if there is any residual Boraflex at any particular Boraflex panel location. Without that information, any residual Boraflex cannot be considered as providing margin. Therefore, the licensee's analysis does not rely on Boraflex.

The licensee's nuclear criticality safety (NCS) analysis describes the methodology and analytical models used to show that the SFP storage rack maximum k_{eff} will be less than 1.0

when flooded with unborated water for normal conditions, and less than or equal to 0.95 when flooded with borated water for normal and credible accident conditions at a 95-percent probability, 95-percent confidence level. For the NFSA, the analysis shows that NFSA rack maximum k_{eff} will be less than or equal to 0.95 when the NFSA is flooded with unborated water at a 95-percent probability, 95-percent confidence level and will be less than or equal to 0.98 if the NFSA is flooded with low density water (i.e., at optimum moderation conditions) at a 95-percent probability, 95-percent confidence level.

3.2 Proposed Changes to NCS Analyses and Fuel Storage Requirements

The proposed TS changes either impact the NCS analyses or implement changes in fuel storage requirements.

- The U-235 enrichment limit would be increased from 4.6 w/% U-235 to 5.0 w/% in TSs 4.3.1.1.a and 4.3.1.2.a for the SFP and NFSA, respectively.
- The SFP storage requirements captured in TS 3.7.18 would undergo a wholesale revision. The new storage requirements would account for: (1) the “checkerboarding” of the fuel with empty cells, (2) the decrease in fuel reactivity inherent in its use in the reactor (i.e. burnup credit), (3) the natural decay of fissile materials in the post irradiated fuel, and (4) the negative reactivity of the Rod Cluster Control Assemblies (RCCAs).
 - New Region 1 would require that fuel assemblies be stored in a new checkerboard arrangement (2-out-of-4 storage), but would have no burnup curve. The new Region 1 requirements take credit for the neutron leakage when next to the SFP wall. There would be restrictions relative to the Region 1/Region 2 interface.
 - New Region 2 would have two new burnup curves and all Region 2 storage locations would be available for fuel assembly storage. One of the new burnup curves would also incorporate assembly cooling time for determining whether a fuel assembly is permitted to be stored in Region 2. The storage of any fuel assembly containing a full length RCCA would be permitted in Region 2 without regard to the new burnup curves.
- The soluble boron concentration listed in TS 4.3.1.1.c that is necessary to maintain the NAPS SFP k_{eff} at or below 0.95 during normal operation would be raised from 350 ppm (parts per million by weight) to 900 ppm.

3.3 Method of Review

This safety evaluation (SE) involves a review of the licensee's NCS analyses for the NAPS NFSA and the SFP, which was provided as Attachment 6 to the May 2, 2017, license amendment request. The review was performed consistent with Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” and Section 9.1.2, “Spent Fuel Storage,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.”

The NRC staff also used an internal memorandum dated August 19, 1998, containing guidance for performing the review of SFP NCS analysis, hereafter referred to as the “Kopp Memo” (NRC Memorandum from L. Kopp to T. Collins, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” August 19, 1998. While the Kopp Memo does not specify a methodology, it does provide some guidance on the

more salient aspects of an NCS analysis, including computer code validation. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), for both borated and unborated fuel storage pools. The Kopp Memo has been used as relevant guidance in the NRC staff's review of light-water reactor SFP NCS analyses since its issuance, and was used in the review of this NAPS analysis.

The NRC staff also used the interim staff guidance (ISG) document DSS-ISG-2010-01 for review of the revised NAPS SFP criticality analyses. The guidance in DSS-ISG-2010-01 is used by the NRC staff to review nuclear criticality safety analyses for the storage of new and spent nuclear fuel as they apply to: (i) future applications for construction and/or operating licenses; and (ii) future applications for license amendments and requests for exemptions from compliance with applicable requirements, that are approved after the date of this ISG.

3.4 SFP NCS Analysis Review

3.4.1 SFP NCS Analysis Method

There is no generic or standard NRC-approved methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the NAPS SFP are described in the criticality analysis which was provided as Attachment 6 to the May 2, 2017, application. Some potential non-conservatisms were identified during the review, but as discussed below, sufficient margin is built into the analysis methodology to offset the potential non-conservatisms. Consequently, the methodology is specific to this analysis and is not appropriate for other applications.

3.4.1.1 Computational Methods

The NAPS SFP NCS analysis considers the decrease in fuel reactivity typically seen in PWRs as the fuel is depleted during reactor operation. This approach is frequently used in PWR NCS analyses and is sometimes referred to as burnup credit (BUC). BUC NCS analysis requires a two-step process. The first step relates to depletion where a computer code simulates the reactor operation to calculate the changes in the fuel composition of the fuel assembly. The second step is modeling of the depleted fuel assembly in the SFP storage racks and the determination of the system k_{eff} . The validation of the computer codes in each step is a significant portion of the analysis. Since the NAPS NCS analysis credits fuel burnup (for spent fuel storage in the new Region 2 of the SFP), it is necessary for the NRC staff to consider validation of the computer codes and data used to calculate burned fuel compositions, and the computer code and data that utilize the burned fuel compositions to calculate k_{eff} for systems with burned fuel.

3.4.1.1.1 Depletion Computer Code Validation

NAPS used the T5-DEPL depletion sequence from SCALE 6.0 to perform its depletion step. Prior to this LAR, T5-DEPL had been used once before in a SFP NCS licensing application submitted to the NRC. Its use in that application was approved, but it did not constitute generic approval and there is still no NRC-approved topical report governing its use. Therefore, the use of T5-DEPL must be justified on a case-by-case basis, and the applicability of previous guidance associated with SFP NCS analyses must be established for the use of T5-DEPL as a depletion code for this specific analysis.

In the previous application, the NRC staff identified two concerns with using the SCALE 6.0 T5-DEPL sequence to determine the burned fuel compositions; the adequacy of convergence of neutron fluxes calculated by KENO V.a, and the adequacy of the validation methodology. In its LAR, NAPS addressed both of these concerns.

Flux convergence is a concern because the deterministic codes historically used for this analysis typically iterate on neutron flux throughout the problem to ensure that the maximum flux difference in any mesh between iterations is acceptably small. To check convergence, confirmatory calculations were run by the licensee with many more neutron histories. In these calculations, the computed burned fuel compositions did not change appreciably with many more neutron histories. Thus, the licensee provided reasonable assurance that the calculated assembly average neutron fluxes are adequately converged for the specific model used. The NAPS burned fuel compositions were calculated using a relatively thin slice of an assembly and burned fuel compositions were calculated on an assembly average basis. Assembly slice average burned fuel compositions have been used and accepted by the NRC staff in many prior burned fuel analyses, and are therefore, acceptable.

For the depletion step, BUC NCS analyses typically involve use of a computer code approved by the NRC for the purposes of performing reactor core simulation analyses. Those computer codes have an NRC SE governing their use, including any necessary limitations and conditions. Additionally, those NRC-approved codes are being used by numerous licensees to perform reactor core analysis, thereby providing a feedback mechanism should significant differences be observed between reactor core analyses and actual reactor core performance. Current NRC guidance in DSS-ISG-2010-01 states, "In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." However, that guidance is premised on the applicant's using their approved core management codes for the depletion analysis. Instead, NAPS used the T5-DEPL depletion sequence from SCALE 6.0 to perform its depletion step.

Since SCALE 6.0 T5-DEPL calculations is not generically approved for use in in SFP burnup credit analyses, the licensee performed additional fuel depletion calculations using industry standard CASMO4 and CASMO5 codes to show that, for the NAPS calculations, equal or higher in-rack k_{eff} values were generated using the T5-DEPL sequence as compared to those generated using CASMO4 or CASMO5. Since use of the T5-DEPL sequence produces consistently higher in-rack reactivity than CASMO4 or CASMO5, application of the Kopp Memo depletion uncertainty guidance is acceptable for this analysis. The NRC staff notes that the work done for NAPS is too narrow to claim broad applicability to spent nuclear fuel at other plants; however, the use of SCALE 6.0 T5-DEPL sequence with the Kopp Memo (uncertainty equal to 5 percent of the reactivity decrement) is considered acceptable for this NAPS amendment request.

The NRC staff finds that the licensee's analyses provide reasonable assurance that previous guidance regarding the depletion validation is applicable to this NAPS LAR. Consistent with the guidance provided in the Kopp Memo, the licensee's analysis has incorporated an uncertainty equal to 5 percent of the reactivity decrement to cover lack of validation of fuel composition calculations. This uncertainty was calculated by the licensee and applied correctly.

3.4.1.1.2 SFP K_{eff} Computer Code Validation

The study used to support validation of k_{eff} calculations using the SCALE 6.0 CSAS5 sequence is documented in Appendix A of Attachment 6 to the licensee's May 2, 2017, letter. The

validation set includes 321 critical configurations that included 27 of the critical experiments from NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," (ADAMS Accession No. ML082880452), and 294 low enrichment uranium fuel pin experiments from the "International Handbook of Evaluated Criticality Safety Benchmark Experiments," (IHECSBE) (NEA/NSC/DOC(95)3 Volumes IV and VI, Organization for Economic Co-operation and Development, Nuclear Energy Agency, September 2014). The trending analysis did evaluate trends in the calculated bias and bias uncertainty, including those associated with variation of experiment temperature and plutonium content.

The SCALE 6.0 CSAS5 sequence validation was performed in a manner consistent with current NRC guidance. The bias and bias uncertainty determined during the validation were applied in an acceptable manner. Therefore, the NRC staff finds the validation acceptable.

3.4.2 SFP and Fuel Storage Racks

3.4.2.1 SFP Water Temperature

NRC guidance provided in the Kopp Memo states the NCS analysis should be done at the temperature corresponding to the highest reactivity. The licensee's analysis was performed with water temperatures of 32 degrees Fahrenheit (°F), 68 °F (base case), 39 °F, 120 °F, and 140 °F. The water densities used were adjusted consistent with the water temperatures being modeled. The effects of water temperature variation on the maximum k_{eff} value were included as bias terms for all regions, both with and without soluble boron. Therefore, the NRC staff finds that the water temperature and density were appropriately considered in the licensee's criticality analysis.

3.4.2.2 SFP Storage Rack Models

The SFP contains 16 storage arrays of different sizes, with a total storage capacity of 1737 fuel assemblies. Each rack array consists of a welded assembly of individual storage cells. The storage cells are comprised of double wall Type 304 stainless steel boxes welded to each other with tie plates to maintain the cell pitch of 10-9/16 inches. Each storage cell has an interior height of 168 inches to ensure that the top nozzle and core components do not extend above the top of the spent fuel rack when the fuel assembly is fully inserted.

3.4.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The nominal dimensions of cell pitch, cell wall thickness, cell inside dimension, wrapper thickness, and tie plate thickness were used in design basis calculations. To provide estimates for uncertainties associated with rack manufacturing tolerances and uncertainties, the licensee performed sensitivity calculations for each region, for each of the permitted storage configurations, as a function of burnup, and without soluble boron in the pool. The NRC staff finds that the uncertainties were conservatively estimated by the licensee, including proper inclusion of the Monte Carlo uncertainties associated with the CSAS5 calculations used to calculate the sensitivities.

3.4.3 Fuel Assembly

3.4.3.1 Bounding Fuel Assembly Design

The fuel assemblies used at NAPS are all Westinghouse 17x17, or similar assemblies manufactured by other suppliers. The licensee described the four different fuel designs it has used in Section 4.1 of the criticality analysis. The licensee evaluated the NFSA and SFP separately with respect to which fuel assembly would be most limiting. In both cases, the licensee ended up developing different hypothetical fuel assemblies for the analysis. These hypothetical fuel assemblies are intended to bound the current fuel while allowing for some future design changes. The licensee provided analyses that demonstrated these hypothetical fuel assemblies reasonably bound the four different fuel designs it has used to date.

As described in Section 4.1 of the licensee's criticality analysis, NAPS has used a variety of burnable absorbers over its operating history, and expects to use similar absorbers during future plant operation. These absorbers include Pyrex, Burnable Poison Rod Assembly (BPRA), Wet Annular Burnable Absorber (WABA), and Integral Fuel Burnable Absorber (IFBA). The burnable absorbers are discussed further in section 3.4.3.3.5 of this SE.

3.4.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The analysis used a standard approach for quantifying the uncertainty in k_{eff} associated with the fuel assembly manufacturing tolerances and uncertainties. The licensee performed sensitivity calculations for various initial enrichment/final burnup points, in each rack and storage configuration, and without soluble boron. Because of a large margin in the borated scenario, the licensee did not determine specific uncertainties for the borated cases; rather, the licensee used the uncertainties from the unborated analysis. The NRC staff finds that the uncertainty analysis performed by the licensee is thorough, and that the margin in the borated scenario is more than sufficient to justify not determining specific uncertainties for the borated cases, and is, therefore, acceptable.

3.4.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on U-235 enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values would also apply to the spent nuclear fuel. Common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. As mentioned above, the licensee has calculated burnup dependent uncertainties. The characterization of spent nuclear fuel is complex. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: depletion uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup. These characteristics are evaluated in the following sections.

3.4.3.3.1 Depletion Uncertainty

The licensee's determination of the depletion uncertainty is discussed in Section 3.4.1.1.1 of this SE.

3.4.3.3.2 Axial Apportionment of the Burnup or Axial Burnup Profile

Another important aspect of fuel characterization is the selection of the axial burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the ends of the fuel assembly, burnup is suppressed due to neutron leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis" (ADAMS Accession No. ML031110292), has shown that, at assembly burnups above about 10 to 20 gigawatt-days per metric ton of uranium (GWd/MTU), the use of a uniform axial burnup profile results in an under prediction of k_{eff} ; generally the under prediction becomes larger as burnup increases. This is what is known as the "end effect." Proper selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect.

Consistent with the guidance provided in DSS-ISG-2010-01, the NAPS SFP criticality analysis used the bounding axial burnup profiles from NUREG/CR-6801 and uniform profiles where appropriate.

NAPS has never used fuel with axial blankets. Future fuel with axial blankets should be bounded by this analysis provided no credit is taken for the axial blankets. Taking credit for axial blankets would be considered to be a change in methodology.

Consequently, the NRC staff finds that the treatment of axial burnup distribution by the licensee is acceptable.

3.4.3.3.3 Radial Burnup Distribution

Due to the neutron flux gradients in the reactor core, assemblies can show a radially tilted burnup distribution (i.e., differences in burnup between portions or quadrants of the cross section of the assembly). Section 10.1.1 of the NAPS analysis presents a simple analysis to estimate the effect of planar burnup distribution on reactivity. This analysis is limited in that it is based on normal conditions and focuses on a limited set of possible conditions. However, other analyses with similar configurations, such as NUREG/CR-6800, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup Credit Cask Designs," shows comparable reactivity impacts. The licensee has provided 0.01 Δk for margin within the analysis and the NRC staff considers a portion of that sufficient to accommodate the potential non-conservatism that may exist due to the lack of a comprehensive evaluation of the impact of the radial burnup distribution for all possible conditions.

3.4.3.3.4 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (ADAMS Accession No. ML003688150) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on NCS analysis in storage and transportation casks, the basic principles with respect to the depletion analysis apply generically, since the phenomena occur in the reactor as the fuel is being depleted. The results have some applicability to NAPS NCS analyses. The basic strategy for this type of analysis is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum

plutonium-239/241 production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in increased plutonium-239/241 production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. The largest effect appears to be due to moderator temperature. NUREG/CR-6665 approximates the moderator temperature effect, in an infinite lattice of high burnup fuel, to be 90 percent mill per degree Kelvin ($^{\circ}\text{K}$). Thus, a 10°F change in moderator temperature used in the depletion analysis would result in approximately $0.005 \Delta k$. The effects of each core operating parameter typically have a burnup or time dependency.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize plutonium-239/241 production. For fuel and moderator temperatures, the NAPS analysis used a fuel assembly power census based on past operation to calculate moderator and fuel temperatures based on conservative estimates of assembly power (the details of this analysis are found in Section 8 of the licensee's criticality analysis). This is acceptable to the NRC staff, but constitutes a limit on the analysis that must be verified before fuel assemblies are moved from the reactor into Region 2 of the SFP. If this limit is exceeded, the licensee must perform an evaluation pursuant to 10 CFR 50.59 considering the methodology approved in this license amendment to determine if sufficient margin is available. As part of this 10 CFR 50.59 evaluation, the licensee should address the applicability of the validation (Section 6 of the NCS analysis), the impacts on calculated biases or uncertainties, and any new modeling approaches that are not part of the reviewed methodology, to ensure that the margin accepted by the NRC is preserved.

For boron concentration, NUREG/CR-6665 recommends using a conservatively high cycle-average boron concentration. The recommendation to use a cycle-average soluble boron as a conservative modeling assumption is based on the premise that the end of cycle soluble boron concentration is zero, or close to zero. When using a cycle-average soluble boron concentration, the model is actually non-conservative in estimating the plutonium-239/241 production during the first part of the cycle. But toward the end of the cycle, the model conservatively estimates plutonium-239/241 production. When the cycles are completed, the end result is generally accepted to be conservative. However, using a cycle-average soluble boron concentration for "shortened" cycles may be non-conservative in some cases, in that the non-conservative estimation of plutonium-239/241 production from the first part of the cycle may not be balanced by the conservative estimation of plutonium-239/241 production from the latter part of the cycle.

The licensee's analysis used a cycle-average soluble boron concentration of 1100 ppm for all cycles. The largest cycle-average soluble boron concentration in recent operation at NAPS is 1051 ppm. However, the 1051 ppm cycle-average soluble boron concentration came from a shortened cycle. Cycle 22 for NAPS Unit 1 was a shortened cycle, due to a nearby earthquake on August 23, 2011. Two cycles (21 and 22) for NAPS Unit 2 were also shorter due to the earthquake; both of these had cycle-average soluble boron concentrations of 996 ppm. Due to their shorter length, the NRC staff evaluated the details of these three cycles individually. Since these cycles were completed several years ago, the NRC staff was able to consider the actual operating history of all of the fuel assemblies used in those reactor cores, as none of those assemblies are in service any longer. Taking into account the legacy cycle-average soluble boron concentration for the cycles preceding and following the shortened cycles, and considering the ample margin in the analysis, the NRC staff considers these cycles to be

bounded by the analysis. Therefore, the NRC staff finds that the licensee's analysis using 1100 ppm as a cycle-average soluble boron concentration for completed cycles at NAPS is conservative.

3.4.3.3.5 Integral and Fixed Burnable Absorbers

The licensee discussed its analysis of its burnable absorbers in Section 8.9.1 of its criticality analysis. NAPS has used a variety of burnable absorbers over its operating history, and plans to continue to do so in future operations; Pyrex, BPRA, WABA, and IFBA. The Pyrex, BPRA, and WABA are considered "fixed" burnable absorbers as they are "fixed" in place during operation. They can be removed during refueling or other fuel handling activities when the licensee otherwise has access to the fuel assembly. The IFBA is "integral" to the fuel assembly as the poison is part of the fuel assembly itself.

For the analysis of the fixed burnable absorbers, the licensee modeled a BPRA with 24 rods and 3.0 weight percent (w/o) of B_4C . The licensee provided a reasonable justification for assuming the BPRA bound the other fixed burnable absorbers for the reactive effect on the post-irradiated fuel. The justification rests on the BPRA having a higher B-10 content and displacing more water. The Pyrex fixed burnable absorber is a legacy item that won't be used again. Therefore it can be evaluated without regard to potential future changes. The WABA assemblies may be used in the future. In its justification, the licensee did not necessarily consider the maximum boron-10 (B-10) content loading available. The NRC staff's review was limited to the assumptions the licensee used in its analysis. Therefore, the WABA modeled in the licensee's analysis sets an upper bound on the WABA loading for this license amendment. If this bound is exceeded, the licensee must perform an evaluation pursuant to 10 CFR 50.59 considering the methodology approved in this license amendment to determine if sufficient margin is available. As part of this 10 CFR 50.59 evaluation, the licensee should address the applicability of the validation (Section 6 of the NCS analysis), the impacts on calculated biases or uncertainties, and any new modeling approaches that are not part of the reviewed methodology, to ensure that the margin accepted by the NRC is preserved.

The licensee provided an analysis that its maximum IFBA usage to date (200 IFBA rods at 1.5x B-10 loading) plus 6 source fingers is bounded by the 24 BPRA modeled usage. However, this is not the maximum potential IFBA loading. The NRC staff's review was again limited to the assumptions the licensee used in its analysis; therefore, the IFBA modeled in the licensee's analysis sets an upper bound on the IFBA loading for this license amendment. If this bound is exceeded, the licensee must perform an evaluation pursuant to 10 CFR 50.59 considering the methodology approved in this license amendment to determine if sufficient margin is available. As part of this 10 CFR 50.59 evaluation, the licensee should address the applicability of the validation (Section 6 of the NCS analysis), the impacts on calculated biases or uncertainties, and any new modeling approaches that are not part of the reviewed methodology, to ensure that the margin accepted by the NRC is preserved.

The most reactive effect of burnable absorbers occurs when both fixed and integral types are used simultaneously during operations. To address this scenario, the licensee modeled its maximum IFBA usage to date (200 IFBA rods at 1.5x B-10 loading) with an 8 rod BPRA. The licensee provided reasonable assurance this loading would meet the regulatory requirements. This becomes the upper bound on simultaneous fixed and integral burnable absorber usage at NAPS. If this bound is exceeded, the licensee must perform an evaluation pursuant to 10 CFR 50.59 considering the methodology approved in this license amendment to determine if sufficient margin is available. As part of this 10 CFR 50.59 evaluation, the licensee should

address the applicability of the validation (Section 6 of the NCS analysis), the impacts on calculated biases or uncertainties, and any new modeling approaches that are not part of the reviewed methodology, to ensure that the margin accepted by the NRC is preserved.

Dominion has not previously used fuel rods in which gadolinia (i.e., Gd₂O₃) is mixed in with the UO₂ as an integral burnable absorber at NAPS. The licensee's analysis attempts to include gadolinia as an acceptable burnable absorber for NAPS burnup credit by listing previous work. However, the licensee did not provide justification as to why the previous work would be applicable to NAPS, nor did the licensee provide a site specific analysis to include gadolinia as a burnable absorber. The control the licensee proposed for using gadolinia as a burnable absorber is not consistent with the previous work. Therefore, the NRC staff does not consider the use of gadolinia integral burnable absorbers to be within the burnup credit scope of this license amendment. For storage cells where burnup credit is not utilized, when all other conditions are held constant, gadolinia will necessarily reduce the reactivity of the fuel. Therefore, gadolinia is not precluded from fuel assemblies located in storage cells which do not require burnup credit.

3.4.3.3.6 Control Rod Usage

If control rods in RCCAs are inserted for significant amounts of time in the reactor, the associated spectral hardening can increase plutonium generation, leading to higher fuel reactivity for the same burnup. The NAPS analysis identified the average NAPS control rod history for the most recent cycles. The analysis determined that the amount of control rod insertion over the past several cycles was insignificant from a post-irradiation reactivity perspective. Using the cycle average control rod insertion history is not typically an accurate predictor of the impact of control rod insertion on the post-irradiated reactivity of fuel assemblies, because insertion at the end of the cycle will have a larger impact than insertion at the beginning of cycle. However, BPRA and RCCA insertion are mutually exclusive, so the licensee has acceptably evaluated this issue by modelling the BPRA as being present throughout the fuel assembly's entire usage in the reactor. The NRC staff's review of this phenomenon was limited to the assumptions the licensee used in its analysis. Therefore, the NRC staff accepts the licensee's determination that the NAPS cycle average control rod insertion history of two steps has an insignificant impact on reactivity, and that history sets an upper bound on RCCA insertion for this license amendment. If this bound is exceeded, the licensee must perform an evaluation pursuant to 10 CFR 50.59 considering the methodology approved in this license amendment to determine if sufficient margin is available. As part of this 10 CFR 50.59 evaluation, the licensee should address the applicability of the validation (Section 6 of the NCS analysis), the impacts on calculated biases or uncertainties, and any new modeling approaches that are not part of the reviewed methodology, to ensure that the margin accepted by the NRC is preserved.

3.4.3.3.7 Credited Nuclides

The licensee provided a list of nuclides used in the NAPS analysis. The list includes volatile and gaseous fission products. These nuclides may migrate out of the fuel pellets and into the plenum in the fuel rod. The licensee considered the potential for these nuclides to migrate in a manner consistent with how fission gas releases that may occur under severe accident conditions are evaluated. Release fractions used are shown in Table 6.3. These release fractions are conservative for SFP criticality analysis because the SFP accident conditions are not nearly as severe as those considered in the source reference. The NRC staff finds that the accident release fractions used in the licensee's analysis are acceptable.

3.4.4 Non-Standard Fuel Configurations/Reconstituted Fuel

The licensee has provided a description of its fuel reconstitution process. As part of this process, the fuel assembly being reconstituted will be located in a Region 1 storage location, with no fuel assembly face adjacent. The licensee proposed a set of five rules (categories) for the storage of post-reconstituted and non-standard fuel in Section 12.6 of its criticality safety analysis. The NRC staff finds the first four rules for storing non-standard fuel assemblies to be acceptable. The fifth rule would allow storage of non-standard assemblies with missing pins on the exterior two rows of the fuel assembly to be stored as normal fuel. This rule is based on the information in NUREG/CR-6835, "Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks," (ADAMS Accession No. ML032880058). However, the analysis in NUREG/CR-6835, while performed with a 17x17 fuel assembly design similar to that used at NAPS, modeled a spent fuel storage cask with a strong neutron-absorbing material (NAM) in the cell walls. Such an analysis based on the model of a cask with a strong NAM is not necessarily applicable to a SFP without a strong NAM. Dominion recognized this fact when it performed a site specific analysis for fuel assemblies with missing interior pins, and placed appropriate limitations on the storage of those assemblies (proposed third rule). Therefore, the NRC staff does not consider the proposed fifth rule in Section 12.6 of the licensee's analysis to be acceptable for this application, as the results from NUREG/CR-6835 have not been demonstrated to be applicable for the analysis of NAPS fuel assemblies with missing exterior pins for this amendment. The NRC staff notes that the licensee identified 3 assemblies currently stored in the NAPS SFP in this category. Absent a site-specific analysis supporting the proposed fifth rule for storing fuel assemblies with missing exterior pins as normal fuel, the NRC considers it acceptable for the licensee to use the proposed third rule for storing all fuel assemblies with missing pins.

In addition to the reconstituted fuel assemblies, NAPS has two "Fuel Rod Storage Racks" (FRSR) that can contain up to 52 fuel rods. The FRSR fit into and are stored in the SFP storage cells. The licensee performed analysis that demonstrated the FRSR can be stored in any location in Region 1 or Region 2.

Additionally, NAPS has several non-fuel items that are stored in the SFP. These items do not contain any fuel and can be stored in any fuel location, consistent with the approved NCS analysis. Similarly, they cannot be stored in any storage cell required to be empty.

3.4.5 Determination of Soluble Boron Requirements for Normal and Accident Conditions

Section 50.68 of 10 CFR requires that the k_{eff} of the NAPS racks, loaded with fuel of the maximum fuel assembly reactivity, must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with borated water. This requirement applies to all normal and abnormal/accident conditions.

The licensee's analysis demonstrated that the k_{eff} for each Region was equal to or less than 0.95 at a 95-percent probability, 95-percent confidence level for normal static conditions with 900 ppm of soluble boron. The licensee's analysis considered normal non-static conditions such as fuel handling and inspection. The licensee's analysis considered that fuel assemblies being moved could come within twelve inches of other fuel assemblies. That proximity could challenge the sub-criticality requirements. Rather than explicitly evaluating the different possibilities, the licensee is imposing a procedural restriction on fuel movement intended to prevent moving fuel assemblies from coming within twelve inches of another fuel assembly.

The NRC staff considers it an implementation requirement that this procedural limit be incorporated and the fuel handler training include how the proximity to other fuel will be controlled.

The licensee's determination of the soluble boron requirements under normal and abnormal/accident conditions is consistent with current guidance, and sufficiently conservative to provide reasonable assurance that the regulatory requirements will be met.

Acceptability of the SFP boron dilution event is based on the ability of the licensee to detect and terminate the event before reaching the soluble boron required to maintain k_{eff} less than 0.95 under normal conditions. The double contingency principle allows independent events to be evaluated separately, therefore it is the soluble boron requirement under normal conditions which is used as the end state for the boron dilution analysis rather than a dilution to zero soluble boron or in conjunction with another event. For this amendment, the licensee did not perform a new boron dilution analysis, instead relying upon the previous boron dilution analysis as bounding the new condition. The licensee has stated that there have been no changes to the plant which would contribute to the boron dilution event. The NRC staff reviewed the previous license amendment (ADAMS Accession No. ML011700557) to compare the description of the previous boron dilution analysis with the information the licensee provided in this LAR. According to the previous amendment, the licensee needed 350 ppm of soluble boron to maintain the SFP k_{eff} less than or equal to 0.95 under normal conditions. The licensee needed another 550 ppm of soluble boron to maintain the SFP k_{eff} less than or equal to 0.95 under accident conditions. For the boron dilution analysis, the licensee added 300 ppm to the 900 ppm needed for the accident condition and assumed the SFP soluble boron was 2300 ppm instead of the TS required 2500 ppm. This was a conservative boron dilution range (2300 ppm diluted to 1200 ppm) over what could reasonably have been used (2500 ppm diluted to 350 ppm). This LAR states under the new conditions the NAPS SFP will require 900 ppm of soluble boron to maintain the SFP k_{eff} less than or equal to 0.95 under normal conditions and that the NAPS TS currently require 2600 ppm of soluble boron in the SFP. The boron dilution range from the previous analysis (2300 ppm diluted to 1200 ppm) is still conservative with regard to what could reasonably be used (2600 ppm diluted to 900 ppm) for this analysis. Absent changes to the plant that could affect the rate of boron dilution, the previous boron dilution analysis would bound the proposed SFP storage requirements. Therefore, the NRC staff accepts the licensee's disposition of the boron dilution event.

The multiple misloading event was determined to be the limiting accident for the higher enrichment fuel. For the Region 1 and Region 2 multiple misloading analysis, the licensee modeled a 6x6 array of fresh 5 w/% U-235 fuel assemblies without burnable poisons with the center 4x4 array placed asymmetrically. The licensee modeled the multiple misload at the SFP TS soluble boron requirement of 2600 ppm. With the exception of the temperature bias, the licensee used previously determined biases and uncertainties. The licensee's reported k_{eff} at a 95-percent probability, 95-percent confidence level, is 0.8698 for this event, including 0.02 Δk margin for NRC review. The licensee provided additional analyses that demonstrated other potential accidents, including a dropped or misplaced assembly, or a SFP over-temperature condition, are bounded by the multiple misloading accident for the 5 w/% U-235 fuel assemblies. The NRC staff reviewed the licensee's evaluation of potential accidents for the proposed fuel design and finds it acceptable.

3.5 New Fuel Storage Area Nuclear Criticality Safety Analysis

Section 7 of the NCS analysis covers new fuel storage in the NFSA. This section documents the review of the NFSA NCS analysis.

3.5.1 NFSA NCS Analysis Method

The SCALE 6.0 CSAS5 KENO V.a-based criticality analysis sequence and the SCALE ENDF/B-VII 238 neutron energy group library were used in the licensee's analysis to calculate the k_{eff} value for fresh fuel in the NFSA fuel storage racks. The CSAS5 sequence has a long history of use for this type of analysis and is, therefore, acceptable. The CSAS5 sequence and the ENDF/B-VII 238 group library validation study is presented in Appendix A of the criticality analysis (Attachment 40 to licensee's May 2, 2017, letter).

The NRC staff reviewed the computational method and supporting validation described above and, based on the results, finds them acceptable.

3.5.2 NFSA Fuel Storage Racks

The steel structures that comprise the NFSA fuel storage racks were modeled. The modeling of the rack structure itself isn't commonly done, but is acceptable provided the analysis includes any bias and/or uncertainty attributable to the rack structure. The licensee included the bias and uncertainty attributable to the rack structure.

The licensee included U^{234} but not U^{236} as part of the fresh fuel. The justification for including U^{234} is reasonable.

The concrete walls surrounding the NFSA can be a significant reflector, especially in the optimum moderation scenarios. The licensee derived and justified an appropriate concrete composition.

The licensee included a temperature bias and asymmetric positioning in the summation of the biases and uncertainties regarding the NFSA.

Compliance with 10 CFR 50.68(b)(2) requires that k_{eff} of the NFSR not exceed 0.95, at a 95 percent probability, 95 percent confidence level assuming the NFSR is flooded with full density water. The licensee determined the k_{eff} of its NFSA to be 0.9230 if it were flooded with full density water. That included 0.01 Δk as margin within the analysis for NRC review. The NRC relied upon that margin to take a graded approach in its review.

Compliance with 10 CFR 50.68(b)(3) requires that k_{eff} of the NFSR not exceed 0.98, at a 95 percent probability, 95 percent confidence level assuming the NFSR is flooded with an optimum density moderator. The licensee determined the k_{eff} of its NFSA to be 0.9748 if it were flooded with an optimum density moderator. That included 0.01 Δk as margin within the analysis for NRC review. The NRC staff relied upon that margin to take a graded approach in its review.

The licensee's analysis demonstrates that the k_{eff} values, including bias and uncertainties, for both full density water and optimum moderation conditions are sufficiently below the applicable limits of 10 CFR 50.68. Therefore, the NRC staff finds that the change to the NFSA is acceptable.

3.6 NRC Staff Conclusion

The NRC staff has completed its review of the NAPS SFP and NFSA NCS analyses, which are documented in the licensee's May 2, 2017, letter and concludes that there is reasonable assurance that the NAPS SFP and NFSA fuel storage racks meet the applicable regulatory requirements in 10 CFR 50.68 for storage of the specified fuel assembly designs.

In addition, the NRC staff concludes that conservatisms and potential non-conservatisms have been adequately addressed in the analysis via licensee margins and confirms compliance with NRC regulatory limits.

The analysis and methodology used in the licensee's application is unique to NAPS SFP and NFSA and, therefore, is solely applicable to NAPS. There are several aspects about the analysis and methodology that will make future changes under the auspices of 10 CFR 50.59 complex. For example, the licensee demonstrated an acceptable use of the SCALE TRITON depletion sequence for this application. However, there is no regulatory guidance on the use of the SCALE TRITON depletion sequence and the NRC staff would consider any change in the way SCALE TRITON depletion sequence was used in this analysis a deviation from the approved methodology. Additionally, the NRC's evaluation included consideration of offsetting effects and is therefore specific to and applicable to NAPS only. The licensee should ensure full and appropriate consideration of these offsetting effects when evaluating potential changes to the facility and methods of evaluation that may impact these criticality analyses.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Virginia official was notified of the proposed issuance of the amendments on May 16, 2018. On May 16, 2018, the official confirmed that the Commonwealth had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 6, 2018 (83 FR 9553). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: K. Wood

Date: July 27, 2018

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS REGARDING NEW AND SPENT FUEL STORAGE (CAC NOS. MF9712 AND MF9713; EPID L-2017-LLA-0240) DATED July 27, 2018

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