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

June 16, 2015

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

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS TO EXTEND TYPE A TEST FREQUENCY TO 15 YEARS (TAC NOS. MF4332 AND MF4333)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 274 and 256 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 30, 2014, as supplemented by letter dated January 28, 2015.

These amendments revise Technical Specification 5.5.15, "Containment Leakage Rate Testing Program". Specifically, this will replace the reference to Regulatory Guide 1.163 with the Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI 94-01, Revision 2" of the Safety Evaluation in NEI 94-01, Revision 2-A, dated October 2008, as the implementation document.

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

Sincerely,

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

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 274 to NPF-4
2. Amendment No. 256 to NPF-7
3. Safety Evaluation

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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. 274
Renewed License No. NPF-4

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

   A. The application for amendment to North Anna Power Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-4 filed by the Virginia Electric and Power Company, (the licensee) dated June 30, 2014, as supplemented by letter dated January 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 274 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: June 16, 2015

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. 256
Renewed License No. NPF-7

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

   A. The application for amendment to North Anna Power Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-7 filed by the Virginia Electric and Power Company, (the licensee) dated June 30, 2014, as supplemented by letter dated January 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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

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

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 256 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: June 16, 2015

ATTACHMENT TO

LICENSE AMENDMENT NO. 274

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 256

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

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

Remove Pages	Insert Pages
Licenses	Licenses
License No. NPF-4, page 3	License No. NPF-4, page 3
License No. NPF-7, page 3	License No. NPF-7, page 3
TSs	TSs
Remove	Insert
5.5	5.5

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;

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

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

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 274are 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. 256 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

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI TR 94-01, Revision 2" of the NRC Safety Evaluation Report in NEI 94-01, Revision 2A, dated October 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.7 psig. The containment design pressure is 45 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 274 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 256 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 application dated June 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14183B318) as supplemented by letter dated January 28, 2015 (ADAMS Accession No. ML15034a353), the Virginia Electric and Power Company (the licensee) requested changes to the Technical Specification (TS) for Facility Operating License Nos. NPF-4 and NPF-7 for North Anna Power Station (NAPS), Units 1 and 2, respectively.

The proposed changes would revise TS 5.5.15, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163 (September 1995) with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A (ADAMS Accession No. ML122210254) and Section 4.1 "Limitations and Conditions for NEI 94-01, Revision 2" of the U.S. Nuclear Regulatory Commission (NRC) Safety Evaluation in NEI 94-01, Revision 2-A, (Reference 7.4) dated October 2008, as the implementation document to develop the 10 CFR 50, Appendix J, Option B, performance-based primary containment leakage testing program. This amendment would allow NAPS, Units 1 and 2, to extend its performance-based primary containment integrated leakage rate test (ILRT), i.e. Type A test, interval to 15 years and the primary containment isolation valve local leakage rate test (LLRT), i.e. Type C test interval to 75 months. Accordingly, the next Type A test at NAPS, Unit 1, would be conducted by October 11, 2022, instead of the current due date of October 11, 2017. The last NAPS, Unit 2, Type A test which was due by October 9, 2014, was completed on October 4, 2014. The last NAPS, Unit 1 and 2, Type C tests which were due in 60 months interval, were completed on March 21, 2015 and October 2, 2014 respectively.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.54(o) require that the primary containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR Part 50 includes two options, Option A – Prescriptive Requirements, and Option B - Performance-Based Requirements, either of which can be chosen for meeting the requirements of Appendix J.

The testing requirements in Appendix J ensure that leakage through the primary containment and related systems and components penetrating primary containment do not exceed allowable leakage rate values specified in the TSs or associated bases; and integrity of the containment structure is maintained during its service life.

The licensee has adopted and has been implementing Option B for meeting the requirements of Appendix J. Option B of Appendix J specifies the performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate; Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries such as penetrations; and Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the allowable leakage rate with margin, as specified in the Technical Specifications. Option B also requires that a general visual inspection for structural deterioration of the accessible interior and exterior surfaces of the containment, which may affect the containment leak-tight integrity, be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

Section V.B.3 of 10 CFR 50 Appendix J, Option B, requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage-testing program be included, by general reference, in the plant technical specifications. Furthermore, the submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

The implementation guidance document that is currently referenced in the NAPS, Units 1 and 2, TS 5.5.15, "Containment Building Leakage Rate Testing Program," is RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 endorsed TR NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995, as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B to Appendix J to 10 CFR Part 50, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

TR NEI 94-01, Revision 2-A, describes an approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J. It incorporates the regulatory positions stated in RG 1.163, and includes provisions for extending Type A test intervals to up to 15 years. In the NRC safety evaluation (SE), dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1 of the safety evaluation.

Guidance for extending Type C LLRT intervals beyond 60 months up to 75 months is given in TR NEI 94-01, Revision 3-A (Reference 3).

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes

NAPS, Units 1 and 2, TS 5.5.15, "Containment Leakage Rate Testing Program," currently states in part,

> A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, modified by the following exception: NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 9, 1999 Type A test shall be performed no later than October 9, 2014.

The LAR proposed to replace the reference to RG 1.163 with a reference to NEI 94-01 Revisions 2-A and 3-A. The proposed change will revise TS 5.5.15 to state, in part:

> A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI TR 94-01, Revision 2" of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

3.2 Adoption of Topical Report NEI 94-01, Revision 3-A

As described in NRC letter dated August 20, 2013, to NEI "Request Revision to Topical Report NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML13192A394), Revision 3-A, inadvertently did not include the six limitations and conditions in NRC's SE (ADAMS Accession No. ML081140105) approving NEI 94-01, Revision 2. Although the six limitations and

conditions were not included in NEI 94-01, Revision 3-A, they apply to a licensee's request to use NEI 94-01, Revision 3-A, requesting to extend the ILRT.

The NRC staff primary review method was to ensure the six limitations and conditions as set forth in the SE (ADAMS Accession No. ML081140105), endorsing NEI 94-01, Revision 2, were met. In the staff SE, the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of Appendix J to 10 CFR Part 50, and is acceptable for referencing by licensees proposing to amend their TS regarding containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1. Section 3.1 of the staff SE provides the NRC staff position on the adequacy of NEI 94-01 in addressing the performance-based Type A, Type B, and Type C test frequencies. It also addresses the adequacy of pre-test inspections, procedures to be used after major modifications to the containment structure, deferral of tests beyond the 15-year interval, and the relation of containment in-service inspection (CISI) requirements mandated by 10 CFR 50.55a to the containment leak rate testing requirement.

The NRC staff evaluated whether the licensee adequately addressed and satisfied the six limitations and conditions as set forth in the SE (ADAMS Accession No. ML081140105) dated June 25, 2008, as discussed below.

3.2.1 NRC Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002.

The licensee stated in Section 4.0 of the LAR (Reference 1) that NAPS, Units 1 and 2, will use the definition in Section 5.0 of NEI 94-01, Revision 3-A, for calculating the Type A leakage rate when future Type A tests are performed on a continuing compliance basis. Since the licensee has committed to comply with the definition in NEI 94-01, Revision 3-A (NEI 94-01, Revision 3-A contains the same definition as in Revision 2-A) for calculating the Type A test leakage rate, the NRC staff finds that the licensee has adequately addressed Condition 1 in the NRC staff SE for NEI 94-01, Revision 2.

3.2.2 NRC Condition 2

> The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.

NEI 94-01, Section 9.2.3.2, "Supplemental Inspection Requirements," states that in order to provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval of the Type A test is extended to 15 years.

The licensee stated that (1) the examinations performed in accordance with the NAPS, Units 1 and 2, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel

Code (Code), Section XI, Subsections IWE and IWL program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B; (2) a general visual inspection of accessible surfaces of the metallic components of the containment pressure boundary is required during each IWE inspection period. Therefore, ASME Code, Section XI, Subsection IWE assures that at least three general visual examinations of metallic components will be conducted before the next Type A test if the Type A test interval is extended to 15 years; (3) visual examinations of accessible concrete containment components in accordance with ASME Code, Section XI, Subsection IWL are performed every five years, resulting in at least three IWL examinations being performed during a 15-year Type A test interval; and (4) the NAPS Technical Requirements Manual (TRM) surveillance requirements require a general visual examination of the accessible interior and exterior surfaces of the containment prior to initiating a Type A test.

Additionally, the LAR (Reference 1) includes an approximate schedule for general visual examinations of accessible interior and exterior containment surfaces, representative of a typical 15-year period between Type A tests.

The licensee concluded that, together, the subsections IWE and IWL examinations and the NAPS TRM surveillance requirements assure that at least four general visual examinations of accessible interior and exterior containment surfaces will be conducted before the next Type A test if the Type A test interval is extended to 15 years. Therefore, the requirements of Section 9.2.3.2 of NEI 94-01 and Condition 2 in Section 4.1 of the NRC staff SE are met.

On the basis that the licensee's schedule of general visual examinations described above results in at least three examinations between Type A tests and one examination immediately prior to the Type A test for both containment concrete and metallic liner surfaces, the NRC staff concludes that the licensee's inspection schedule plan, noted in the LAR, meets the general visual examination requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A, and 10 CFR Part 50, Appendix J, Option B, and therefore, satisfies Condition 2 in the NRC staff SE for NEI 94-01, Revision 2.

3.2.3 NRC Condition 3

> The licensee addresses the areas of the containment structure potentially subjected to degradation.

In Section 4.0 of the LAR (Reference 1), the licensee stated that general visual examination of accessible interior and exterior surfaces of the containment structure for structural problems is conducted in accordance with the NAPS CISI program which implements the requirements of the ASME Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). Under the NAPS CISI program, as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A), the licensee evaluates the acceptability of inaccessible areas of the containment structure and metallic liner if conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

The licensee, in letter dated January 28, 2015 (Reference 2), in response to the staff RAI, stated that (1) NAPS, Units 1 and 2, do not have moisture barriers located at the interface of the containment floor slab and the containment wall liner; and (2) the NAPS, Unit 1, IWE general

visual examinations have not identified any liner degradation at the interface of the containment floor slab and the containment wall liner. The licensee further stated that during NAPS, Unit 2, fall 1999 refueling outage, evidence of some rust in a number of areas at the interface of the concrete floor slab with the containment wall liner was discovered. Four of these areas were evaluated and the apparent cause was due to water spraying during containment decontamination efforts during outages. The four inaccessible areas were excavated and augmented IWE Category E-C examinations were performed using VT-1 visual and ultrasonic test (UT) methods to measure liner plate thickness of the affected areas. The licensee performed necessary evaluations to ensure the affected liner areas were adequate for their intended design basis function. The licensee, in letter dated April 3, 2008 (ADAMS Accession No. ML080950111), stated that three consecutive UT examinations (March 2001, October 2005 and March 2007 refueling outages) and VT-1 examinations (March 2001, May 2004, and March 2007 refueling outages) were performed and no evidence of continued degradation was observed. The staff notes that this finding was also described in the SE (ADAMS Accession No. ML081510562) dated June 30, 2008, for one-time 5-year extension of Type A test interval for NAPS, Unit 2.

In Section 4.4.1 of the LAR (Reference 1), the licensee stated that the NAPS initial ten-year in-service inspections revealed no Unit 1 areas deemed susceptible to accelerated degradation and aging; therefore, augmented examinations per Category E-C were not required. Three areas in Unit 2 were deemed susceptible to accelerated degradation and aging due to wood entrapment inside the concrete and required augmented examinations per Category E-C. Corrective action to remove the wood was performed and the areas were re-examined in accordance with IWE-2420(b) and after three (3) augmented examinations were performed and remained essentially unchanged in accordance with IWE-2420(c) the inspection frequency was returned to normal.

Based on the above information provided by the licensee in the LAR relative to the NAPS, Units 1 and 2, operating experience, and the information provided in response to staff RAI, no conditions have been identified that would indicate the presence of any significant degraded conditions in accessible or inaccessible areas of the concrete containment structure and steel liner affecting the leak tightness or structural integrity of the NAPS, Units 1 and 2, containment structures. Therefore, the NRC staff concludes that the licensee has adequately addressed the intent of Condition 3 of the NRC staff SE for NEI 94-01, Revision 2.

3.2.4 NRC Condition 4

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable.

The licensee stated in the LAR (Reference 1, Section 4.0), that (1) NAPS, Units 1 and 2, had already replaced the steam generators, and the replacement did not require modifications to the containment structure; (2) when NAPS, Units 1 and 2, replaced the reactor vessel closure head, the containment structure was modified and the design change process addressed the testing requirements of the containment structure modifications.

Furthermore, the licensee stated in the LAR (Reference 1, Section 4.2) that (1) any unplanned modifications to the containment structure prior to the next scheduled Type A test would be

subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J; (2) there have been no pressure or temperature excursions in the containment which could have adversely affected containment integrity; and (3) there is no anticipated addition or removal of plant hardware within the containment structure which could affect leak-tightness.

Based on the above, the NRC staff concludes that the licensee's program will implement the staff position with regard to post-repair pressure testing following major and minor containment repairs and modifications, as explained in Section 3.1.4 of the NRC staff SE for NEI 94-01, Revision 2. Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 4 of the NRC staff SE for NEI 94-01, Revision 2.

3.2.5 NRC Condition 5

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.

The licensee stated, in Section 4 of the LAR, that it acknowledges and accepts the NRC staff position in Condition 5, as communicated to the nuclear industry in NRC Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50," dated December 8, 2008 (ADAMS Accession No. ML080020394).

Accordingly, the NRC staff concludes that the licensee has confirmed its understanding that any extension of the Type A test interval beyond the upperbound performance-based limit of 15 years should be infrequent and should be requested only for compelling reasons, and that the NRC staff will implement the position in RIS 2008-27 in reviewing such LARs. Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 5 of the NRC staff SE for NEI 94-01, Revision 2.

3.2.6 NRC Condition 6

> For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No.1 009325, Revision 2, including the use of past containment ILRT data.

The licensee stated in its LAR that this condition is not applicable to NAPS, Units 1 and 2, since NAPS, Units 1 and 2, are not licensed to 10 CFR Part 52. The NRC staff concludes that Condition 6 does not apply to NAPS, Units 1 and 2, since NAPS, Units 1 and 2, are currently operating reactors licensed to 10 CFR Part 50.

Based on the above evaluation, the NRC staff concludes that the licensee has adequately addressed and satisfied the six conditions in Section 4.1 of the NRC staff SE for NEI 94-01, Revision 2.

3.3 Extension of Type A Test Interval from 10 to 15 Years

As described in Reference 1 and NAPS updated safety analysis report (UFSAR), the NAPS Units 1 and 2, are pressurized water reactors with a steel-lined reinforced concrete containment structure with vertical cylindrical wall and hemispherical dome, supported on a flat foundation base mat. A waterproof membrane was placed below the containment foundation base mat and carried up the containment wall to above ground-water level. The containment wall steel liner is 3/8 inch thick. The steel liner for the base mat consists of a 1/4 inch plate except in the in-core instrumentation area, where an exposed 3/4 inch plate is used, and the inside recirculation spray pump sumps, where an exposed 1/2 inch plate is used. The base mat liner plate is overlaid with a reinforced-concrete slab. The steel liner for the dome is 1/2 inch thick. The steel liner functions primarily as a gas tight membrane. No credit has been taken for the presence of the steel liner in the design of the containment structure to resist seismic forces or other design loads. During power operation, NAPS, Units 1 and 2, are maintained at a sub-atmospheric condition.

The containment pressure boundary consists of the steel liner, containment access penetrations, and penetrations for process piping and electrical wiring. The NAPS, Units 1 and 2, leak-tight integrity of the penetrations and isolation valves are verified through Type B and Type C LLRTs and the overall leak-tight integrity and structural integrity of the primary containment is verified through a Type A test, as required by 10 CFR 50, Appendix J. The leakage rate testing requirements of 10 CFR 50 Appendix J Option B (Type A, Type B and Type C Tests) and the CISI requirements mandated by 10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment during its service life.

3.3.1 Historical Plant-Specific Containment Leakage Testing Program Results

As indicated in the LAR (Reference 1) and the NAPS, Units 1 and 2 TS, the maximum allowable containment leakage rate, L_a, is 0.1 percent of containment air weight per day at the peak calculated containment internal pressure for design basis loss-of-coolant accident, P_a.

In Section 4.2 of the LAR, the licensee provided the NAPS, Units 1 and 2, Type A test results history which were all below the containment leakage rate, L_a. The last Unit 1 Type A test was performed on October 11, 2007 and the as-found leakage rate was 56 percent of the allowable containment leakage rate, L_a. The last Unit 2 Type A test was performed on October 9, 1999, and the as-found leakage rate was below 62 percent of the allowable containment leakage rate, L_a. The current NAPS TS requires the next Type A test, after the October 9, 1999, Type A test, be performed no later than October 9, 2014. This one-time 5-year extension of Type A test interval for NAPS, Unit 2, was approved by staff safety evaluation (ADAMS Accession No. ML081510562) dated June 30, 2008. The LAR was submitted by letter dated June 30, 2014, and did not include the results of 2014 NAPS, Unit 2, Type A test. In response to staff RAI, the licensee, in letter dated January 28, 2015, stated that the 2014 NAPS, Unit 2, Type A test results are less than 40 percent of the NAPS TS limit.

In Section 4.3 of the LAR (Reference 1), the licensee stated that the NAPS, Units 1 and 2, Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and NAPS, Units 1 and 2, TS 5.5.15.

The licensee also provided the NAPS, Units 1 and 2, Type B and C maximum and minimum pathway total leakage rate for the 2010 to 2013 refueling outages. The licensee indicated that (1) the leakage rates were well below the acceptance criterion of combined Type B and Type C leakage rate of 182.6 standard cubic feet per hour; (2) the current total penetration leakage on a minimum path basis is less than 10 percent of the leakage allowed for containment integrity; and (3) there were no Type B or Type C penetration test failures during the 2013 NAPS, Units 1 and 2, refueling outages.

The licensee stated that industry experience has shown that the Type B and C tests can identify the vast majority (greater than 95%) of all potential primary containment leakage paths. The licensee stated that this LAR adopts the guidance in NEI 94-01, Revision 3-A, in place of NEI 94-01, Revision 0, but otherwise does not affect the scope or performance of Type B or Type C tests, and that Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

In supplemental letter dated January 28, 2015 (ADAMS Accession No. ML15034A353), the licensee provided the percent of the total number of Type B and Type C tested components that are on 120-month and 60-month extended performance-based test interval, as follows:

a) With the exception of one component, all NAPS, Unit 1, Type B tested electrical penetrations are on the extended 120-month test interval.

b) All NAPS, Unit 2, Type B tested electrical penetrations are on the extended 120-month test interval.

c) The air locks and the fuel transfer tube are tested every refueling outage.

d) Although 92 percent of NAPS, Units 1 and 2, Type C tested isolation valves perform well enough to be on an extended test interval, only approximately 40% are tested at that frequency. Because of scheduled maintenance and testing methods, Type C penetrations are routinely tested more frequently than the 60-month interval.

In the LAR, the licensee also provided the NAPS, Units 1 and 2, Type B and C combined as-found minimum pathway total leakage rates. The TS criterion for the combined Type B and C as-found test minimum pathway leakage rate total is 0.6 L_a (182.6 standard cubic feet per hour, scfh). The total combined Type B and Type C as-found minimum pathway leakage rate for Unit 1 recorded in the October 2010, April 2012 and October 2013, refueling outages was less than 5.4% of the TS performance criterion. The total combined Type B and Type C as-found minimum pathway leakage rate for Unit 2 recorded in the April 2010, November 2011 and May 2013, refueling outages was less than 6.5% of the TS performance criterion.

The licensee stated in the LAR that the proposed change from the guidance in NEI 94-01, Revision 3-A, in place of NEI 94-01, Revision 0, would allow for extending Type C test intervals to 75 months, but otherwise does not affect the scope or performance of Type B or Type C

tests, and that Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

Based on a review of the licensee's responses to the RAIs, and based on the information discussed above, the NRC staff concludes that (1) the performance history of successful completion of most recent Type A tests supports extending the current ILRT interval to 15 years; (2) the combined leakage from the Type B and Type C tests has been consistently maintained below the acceptance criteria; (3) a majority of penetrations that are subject to performance based Type B or Type C test are on or could be on the maximum allowed performance based interval of 120 months or 60 months, as applicable, which demonstrates good performance of Type B and Type C penetrations at NAPS, Units 1 and 2; (4) there is reasonable assurance that the licensee is effectively implementing its Type B and Type C testing program under Option B of Appendix J to 10 CFR Part 50.

The NRC staff concludes that the performance history of the more recent Type A tests with a large margin to the performance criterion supports extending the current ILRT interval to 15 years; and that the combined leakage from the Type B and Type C tests have also been consistently maintained well below the acceptance criterion which supports extending the current Type C test intervals to 75 months. Therefore, there is reasonable assurance that the licensee will continue to effectively implement its Type B and Type C testing program under Option B of Appendix J to 10 CFR Part 50 and maintain the leak-tight integrity of the primary containments.

3.3.2 Containment In-Service Inspection Program

In Section 4.0 of the LAR, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment system for structural problems are conducted in accordance with the NAPS, Units 1 and 2, CISI program and schedule, which implements the requirements of the ASME Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The IWE/IWL inspections and supplemental inspections, in accordance with other approved plant procedures, are used to satisfy the general visual examination requirements of Appendix J, Option B and to monitor and manage the age-related degradations of the primary containment to ensure that containment structural and leak-tight integrity is maintained through its service life.

The examination of the liner plate at the interface of the concrete base floor and the containment wall liner plate has been discussed in Section 3.2.3 of this safety evaluation and will not be repeated here.

The licensee provided information regarding the results of IWL inspections and corrective actions taken to disposition them. The licensee stated that the IWL second interval concrete containment examinations have specified dates of August 31, 2011 and August 31, 2016 for Units 1 and 2. General and detailed visual examinations were completed by the required August 31, 2011, date for the first five year period in the summer of 2011 in accordance with Examination Category L-A of the 2001 Edition with 2003 Addenda of ASME Code Section XI. The second 5-year concrete containment examination in accordance with Examination Category L-A of the code is scheduled to be completed in the summer of 2016 for Units 1 and 2. The 2011 examinations on the concrete exterior were conducted by the Responsible Engineer using

the approved ASME Code visual methods. During the examinations, 25 indications were observed on each unit. The Unit 1 and Unit 2 indications noted were minor spalls, efflorescence pop-outs, cracks, stains, and abandoned anchors/anchor holes. Almost all conditions identified were minor in nature and did not require additional excavation for repair. In general, the indications requiring additional inspection or excavation involved embedded materials and loose or hollow-sounding areas. The repairs were designated as cosmetic with the exception of one code repair area for Unit 1. The code repair indication was a spall/rock pocket located on the Unit 1 dome approximately 4 feet long 5 inches wide and 3 inches deep that exposed primary reinforcement. The area was repaired using safety-related repair concrete in accordance with NAPS procedures and a VT-1 examination was performed. In summary, the licensee stated that no significant defects or concerns were observed on the exterior concrete and concluded that the NAPS, Unit 1 and 2, containment structures were in good material condition.

Conclusion

The licensee, stated that (1) in response to NRC Information Notice 2014-07 "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," a new Examination Category E-A, Containment Surfaces, Item No. E1.30 - Moisture Barriers, has been incorporated into the NAPS CISI Program; (2) for NAPS Unit 2, this additional general visual examination was completed and two deficiencies were noted and condition reports were generated to document these discrepancies. A missing test port panel plug used to test the bottom liner welds during initial construction was replaced. Another test port panel plug which had unidentified deposits was cleaned and inspected. Follow-up general visual IWE examinations were completed for these two items and found them in satisfactory conditions; (3) the NAPS, Unit 1, examination is scheduled for the upcoming 2015 refueling outage. However, most of these components (outside the In-core Instrumentation area) received a general visual inspection during the forced NAPS, Unit 1, cold shutdown in December 2014. No deficiencies were noted.

Based on the results of the recent IWE/IWL inspections discussed above, the NRC staff finds that there has not been evidence to date of significant degradation of the NAPS, Units 1 and 2, containment structures, and the degradations noted have been entered into the NAPS Units 1 and 2, corrective action programs, and appropriately managed and/or corrected. The results of the past ILRTs, LLRTs, and the CISI programs demonstrate acceptable performance of the NAPS, Units 1 and 2, containments and demonstrate that the structural and leak-tight integrity of the containment structure is adequately managed. The structural and leak-tight integrity of the NAPS, Units 1 and 2, containments will continue to be periodically monitored and managed by the ILTR, LLRT and CISI programs, if the current ILRT interval is extended from 10 years to 15 years. The containment structural and leak-tight integrity will continue to be maintained through an adequate containment leakage rate testing programs, if the current LLRT interval is extended to 75 months. Therefore, the NRC staff concludes that there is reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained, without undue risk to public health and safety, if the current Type A test interval at NAPS, Units 1 and 2, is extended to 15 years and the Type C test interval extended to 75 months.

The next Type A test at NAPS, Unit 1, may be conducted no later than October 11, 2022, in lieu of the current due date of October 11, 2017. The last NAPS, Unit 2, Type A test which was due by October 9, 2014, has been completed on October 4, 2014. Therefore, the next Type A test

interval at NAPS, Unit 2, may be extended up to 15 years from the completion date of the last NAPS, Unit 2, Type A test. The next Type C tests at NAPS, Unit 1 and 2, may be conducted no later than 75 months interval from the completion date of the last Type C tests performed on the respective containment isolation valves.

3.4. Probabilistic Risk Assessment (PRA)

3.4.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of Nuclear Energy Institute (NEI) 94-01, Revision 3-A (Agencywide Documents Access and Management System (ADAMS) Accession No. ML122210254), "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond ten years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1018243[1], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the safety evaluation report (SER), dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff found the methodology in NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2 of the SER for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submit documentation indicating that the technical adequacy of their Probabilistic Risk Assessment (PRA) is consistent with the requirements of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," relevant to the ILRT extension application. Additional application specific guidance on the technical adequacy of a PRA used to extend ILRT intervals is provided in the SER for EPRI TR-1009325, Revision 2.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6[2] of the SER for EPRI TR-1009325, Revision 2.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided the average leak rate for the pre-existing containment large leak accident case (i.e., accident case

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1 EPRI TR-1018243, is also identified as EPRI TR-1009325, Revision 2-A. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.
2 Section 4.2 of the SER for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

3b) used by licensees is assigned a value of 100 times the maximum allowable leakage rate (L_a) instead of 35 L_a.

4. A LAR is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance. In addition, the change in core damage frequency (CDF) should be calculated and reported.

3.4.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval to once in 15 years. The risk assessment was provided in Attachment 4 to the LAR submitted June 30, 2014 (ADAMS Accession No. ML14183B318). Additional information was provided by the licensee in its letter dated January 28, 2015, in response to NRC RAIs (ADAMS Accession No. ML15034A353).

In Section 4.6.1 of Attachment 1 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 2-A[3] (ADAMS Accession No. ML100620847); the methodology described in EPRI TR-1009325, Revision 2-A; and the NRC regulatory guidance outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Additionally, the licensee used the methodology from Calvert Cliffs Nuclear Power Plant to assess the risk from undetected containment leaks due to steel liner corrosion. In Section 4.4 of Attachment 4 to the LAR and in response to the request for additional information PRA RAI-2, the licensee reviewed the recent steel liner corrosion events and concluded that the corrosion sensitivity study based on Calvert Cliffs methodology included in the LAR remains adequate to represent the current data trends.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SER. A summary of how each condition has been met is provided in the sections below.

3.4.2.1 Technical Adequacy of the PRA

The first condition stipulates that the licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

Consistent with the information provided in Regulatory Issue Summary (RIS) 2007-06 (ADAMS Accession No. ML070650428), "Regulatory Guide 1.200 Implementation," the NRC staff will use Revision 2 of RG 1.200 (ADAMS Accession No. ML090410014) to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. In Section 3.2.4.1 of the SER for NEI 94-01, Revision 2 and EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category (CC) I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for IRLT extension applications, since approximate

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3 NEI 94-01, Revision 3-A (ADAMS Accession No. ML12221A202), added guidance for extending Type C Local Leak Rate Test (LLRT) surveillance intervals beyond sixty months. The guidance for extending Type A ILRT surveillance intervals beyond ten years is the same as that in Revision 2-A.

values of CDF and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

In Attachment 5 to the LAR, the licensee states that the most recent at-power internal events model, referred to as NAPS-R07, dated October 16, 2013, was used to analyze the risk of extending the Type A ILRT interval to 15 years for NAPS, Units 1 and 2. The licensee also states that the PRA model is maintained and updated under a PRA configuration control program in accordance with Dominion procedures. Plant changes, including physical and procedural modifications and changes in performance data, are reviewed and the PRA model is updated to reflect such changes periodically by qualified personnel, with independent reviews and approvals.

The licensee states that the NAPS PRA model received formal industry peer reviews in 2001 and 2013. The licensee states that all facts and observations (F&Os) and their associated dispositions from the 2001 peer review were reviewed by the 2013 peer review team, which concluded that no additional work was needed to address the 2001 F&Os. The 2013 review was a full-scope peer review by the Pressurized Water Reactor Owners Group and it was performed against the requirements of the ASME/ANS RA-Sa-2009 PRA standard and RG 1.200, Revision 2. The 2013 peer review found that 92% of the supporting requirements met or exceeded the requirements of CC I/II. The peer review team identified 72 F&Os, out of which 35 were suggestions, 35 were findings and 2 were best practices. The licensee states that the 35 suggestions do not affect the technical adequacy of the PRA model and have no impact on the results of the ILRT extension evaluation. The licensee provided in Table B-1 of Attachment 5 to the LAR the 35 findings from the 2013 peer review, their associated dispositions and evaluations of their impact on the ILRT extension application. Upon review, the NRC staff requested additional information and the review of these responses and table are discussed below.

In response to the request for additional information (RAI) dated January 28, 2015,(ADAMS Accession No. ML15034A353), the licensee provided a list of 18 F&Os from the 2013 peer review and their associated supporting requirements. These supporting documents have not met CC I of the ASME/ANS RA-Sa-2009 PRA standard as endorsed by RG 1.200, Revision 2. The licensee stated that although each of these F&Os was evaluated individually, their cumulative impact is not expected to exceed the increase in risk evaluated by the bounding sensitivity study. 13 F&Os were related to documentation of assessment and 5 F&Os were related to PRA modeling. The staff evaluated the licensee's assessment of the 13 F&Os that were related to better documentation regarding sources of uncertainty and discussions to meet the supporting requirements. The staff concluded that the licensee has documented the impact on application for each of the F&Os in Attachement 1 of the RAI response. There is no risk impact on CDF or LERF as these are primarily to enhance documentation. As a result, the staff accepts that these 13 F&Os have no impact on the ILRT extension application.

The licensee dispositioned modeling F&O IFSO-A5-01 by performing a sensitivity study and concluded that the CDF impact is insignificant. As specified in response to PRA RAI-1.a and clarified in a conference call with the licensee (ADAMS Accession No. ML15068A090), the licensee stated that the cumulative impact of findings IE-A6-01, DA-B2-01, DA-C14-01 and QU-B5-01 is bounded by an order of magnitude increase in CDF and LERF. In Enclosure 1 of Attachment 5 to the LAR the licensee demonstrated that with this order of magnitude increase in

CDF and LERF the risk acceptance criteria are still met, as discussed in Section 3.4.2.2 of this SE. The NRC staff accepts the licensee's assessment that the 5 open modeling F&Os have no significant impact on the ILRT extension.

In Section 3.2.4.2 of the SER for NEI 94-01, Revision 2 and EPRI TR-1009325, Revision 2, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

The licensee evaluated impact from external events on the ILRT extension in Section 5.7 of Attachment 4 to the LAR, using information from the NAPS Individual Plant Examination of External Events (IPEEE). The licensee's assessment takes into account fire and seismic risks. The licensee states that "other external events such as high winds, external floods, transportation, and nearby facility accidents were considered and screened in the IPEEE," and therefore considered their risk impact to be "negligible compared to the impact associated with internal fires and seismic events."

The licensee states that in the IPEEE analyses, Unit 2 internal fire CDF was higher than the Unit 1 internal fire CDF and, therefore, selected to use the Unit 2 CDF (4.08E-06/year) as a representative fire CDF for both Unit 1 and Unit 2 to assess the risk impact from the ILRT extension. The licensee further states that the IPEEE study for NAPS did not quantify seismic risk. The licensee estimated a seismic CDF of 8.16E-06/year by analogy to Surry seismic CDF. The NRC staff also considered results of an NRC study published in "Results of Safety/Risk Assessment of Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (ADAMS Accession No. ML100270582), which estimated the seismic CDF for NAPS as 4.4E-05/year. Because of sufficient margin between the risk acceptance criteria and the risk associated with ILRT extension at NAPS, as discussed in Section 3.4.2.2 of this SE, the risk of ILRT extension using the higher seismic CDF values from the NRC study still meets the acceptance criteria.

The licensee stated that its PRA was evaluated against the current ASME PRA standard and Revision 2 of RG 1.200, evaluated the findings for applicability to the ILRT interval extension, addressed the findings or explained their impact.

The NRC staff reviewed these findings and accepts that dispositioned findings have been adequately addressed for this application and the cumulative impact of all open findings from the peer reviews is not significant for the ILRT interval extension application. Furthermore, the NRC staff concludes that the impact from external events is appropriately considered. Therefore, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.4.2.2 Estimated Risk Increase

The second condition stipulates that the licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.5 of the NRC SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem/year or 1% of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment over-pressure for net positive suction (NPSH) for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed further in Section 3.4.2.4 of this SE, NAPS credits containment over-pressure. Thus, the associated risk metrics include: CDF, LERF, population dose, and CCFP.

The licensee provided the results of the plant-specific risk assessment in Section 4.6.3 of Attachment 1 to the LAR. Details of the risk assessment are provided in Attachment 4. The risk assessment is applicable to both NAPS, Unit 1 and Unit 2. The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years. The NRC staff's conclusions drawn from the licensee's analysis associated with extending the Type A ILRT frequency are discussed below:

1. The reported increase in LERF for internal events is 1.60E-08/year for both units. These changes are considered to be "very small" (i.e., below 1E-07/year) per the acceptance guidelines in RG 1.174. The reported increase in LERF is 1.29E-07/year for combined internal and external events, for both units. The risk contribution from external events includes the effects of internal fires and seismic events, as discussed in Section 3.4.2.1 of this SE. These changes are considered to be "small" (i.e., between 1E-06/year and 1E-07/year) per the acceptance guidelines in RG 1.174. An assessment of total baseline LERF is also required, to show that the total LERF is less than 1E-05/year. The total base LERF, including internal and external events, is estimated by the licensee to be 1.23E-06/year for both units, which is below 1E-05/year. Because there is sufficient margin to the acceptance criteria, using a higher seismic CDF value consistent with the results of the NRC study discussed in Section 3.4.2.1 of this SE, the change in LERF is still considered as "small" per the acceptance guidelines in RG 1.174.

As discussed in Section 3.4.2.1 of this SE, the licensee performed a bounding sensitivity study (Enclosure 1 of Attachment 5 to the LAR) and demonstrated that the open gaps in NAPS PRA models are bounded by an order of magnitude increase in CDF and LERF for internal events. Assuming an order of magnitude increase in CDF, the licensee calculated an increase in LERF for internal events of 1.59E-07/year for both units, which corresponds to a "small" increase per the acceptance guidelines in RG 1.174. The NRC staff estimated a total base LERF for this sensitivity study as 2.49E-06/year (1.36E-06/year for internal events, 9.67E-07/year for external events, and 1.59E-07/year increase in LERF due to ILRT extension), which is below 1E-05/year.

2. The requested change in Type A ILRT frequency from three in 10 years to once in 15 years results in an increase in the total population dose is 9.11E-04 person-rem/year, or 0.18% of the total population dose for both units. If assuming an order of magnitude increase in CDF to bound the open gaps in PRA quality, as discussed in Section 3.4.2.1 of this SE, the increase in total population dose was calculated by the licensee as 9.41E-03 person-rem/year. In both cases, the reported increase in population dose is below the values provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2. Thus, this increase in the total dose for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years is 0.93% for both units. The same increase in CCFP is obtained if assuming an order of magnitude increase in internal events CDF to bound the open gaps in PRA quality, as discussed in Section 3.4.2.1 of this SE. The increase in CCFP is below the acceptance guidelines in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

4. As discussed in Section 3.4.2.4 of this SER, the licensee demonstrated that although NAPS credits containment over-pressure, there is no increase in CDF.

Based on its review of the risk assessment results, the NRC staff concludes that the increase in LERF is small and consistent with the acceptance guidelines of RG 1.174, and the increase in the total population dose and the magnitude of the change in CCFP for the proposed change are small and supportive of the LAR. The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded as a result of the requested change, and the use of these three quantitative risk metrics described above collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition is met.

3.4.2.3 Leak Rate for the Large Pre-Existing Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in the table in Section 4.6.1 of Attachment 1 to the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the NAPS plant-specific risk assessment. Accordingly, the third condition is met.

3.4.2.4 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment overpressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 5.8 of Attachment 4 to the LAR, the licensee states that NAPS design basis calculations credit containment overpressure to satisfy the NPSH requirements for recirculation spray (RS) and low-head safety injection (LHSI) in recirculation mode during loss of coolant accidents (LOCA). In Attachment A to Attachment 4 of the LAR, as supplemented by the licensee's response to PRA RAI-3, the licensee performed additional accident analyses to address the effects of a containment leak rate of 100 L_a (the EPRI Class 3b contribution) on available NPSH and demonstrated that NPSH would not be lost for any RS or LHSI pumps. As such, using the guidance from EPRI TR-1009325 Revision 2-A and assuming the availability of NPSH for RS or LHSI pumps at 100 L_a, as demonstrated by the licensee, there is no impact on CDF. Accordingly, the fourth condition is met.

4.0 STATE CONSULTATION

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

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. 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 (79 FR 52070, September 2, 2014). 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 concludes that it is acceptable to approve the proposed license amendment to revise TS 5.5.15, "Containment Leakage Rate Testing Program," to adopt NEI 94-01, Revision 3-A, as the implementation document, extend the performance-based Type A leakage test interval to 15 years, and extend the Type C leakage test interval to 75 months.

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

7.0 REFERENCES

1. Letter dated June 30, 2014, from Mark Sartain, Virginia Electric and Power Company, to USNRC regarding Proposed License Amendment Request for Permanent Fifteen-Year Type A Test Interval, North Anna Power Station Units 1 and 2, Docket Nos. 50-338 and 50-339 (ADAMS Accession No. ML14183B318).
2. Letter dated January 28, 2015, from Mark Sartain, Virginia Electric and Power Company, to USNRC regarding Response to Request for Additional Information for Proposed License Amendment Request for Permanent Fifteen-Year Type A Test Interval, North Anna Power Station. Units 1 and 2, Docket Nos. 50-338 and 50-339 (ADAMS Accession No. ML15034A353).
3. Nuclear Energy Institute Topical Report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," July 2012 (ADAMS Accession No. ML12221A202)
4. Nuclear Energy Institute Topical Report NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847)

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Date: June 16, 2015

June 16, 2015

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

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS TO EXTEND TYPE A TEST FREQUENCY TO 15 YEARS (TAC NOS. MF4332 AND MF4333)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 274 and 256 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 30, 2014, as supplemented by letter dated January 28, 2015.

These amendments revise Technical Specification 5.5.15, "Containment Leakage Rate Testing Program". Specifically, this will replace the reference to Regulatory Guide 1.163 with the Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI 94-01, Revision 2" of the Safety Evaluation in NEI 94-01, Revision 2-A, dated October 2008, as the implementation document.

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

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

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 274 to NPF-4
2. Amendment No. 256 to NPF-7
3. Safety Evaluation

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