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

July 16, 2015

Mr. George H. Gellrich, Site Vice President
Exelon Generation Company, LLC
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EXTENSION OF CONTAINMENT LEAKAGE RATE TESTING FREQUENCY (TAC NOS. MF4898 AND MF4899)

Dear Mr. Gellrich:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 310 to Renewed Facility Operating License No. DPR-53 and Amendment No. 288 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (CCNPP), respectively. These amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated September 18, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14265A219), as supplemented by letters dated February 17, 2015 (ADAMS Accession No. ML15051A409), and April 2, 2015 (ADAMS Accession No. ML15097A009).

These amendments revise the CCNPP TS 5.5.16, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163 (September 1995) with a reference to Topical Report (TR) [Nuclear Energy Institute] NEI 94-01, Revision 3-A, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation in NEI 94-01, Revision 2-A, dated October 2008. These references are the implementing documents for the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B, performance-based primary containment leakage testing program for CCNPP. The proposed changes would allow an increase in the Type A test interval from the current 10 years to a maximum of 15 years, and allow an increase in the Type C test interval from the current 60 months to 75 months. The proposed changes would also delete the one time exceptions granted to the Type A test interval and exceptions from the post-modification Type A test when the steam generators at CCNPP were replaced.

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

Sincerely,

Alexander N. Chereskin, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 310 to DPR-53
2. Amendment No. 288 to DPR-69
3. Safety Evaluation

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-317

Amendment No. 310
Renewed License No. DPR-53

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated September 18, 2014, as supplemented by letters dated February 17, 2015, and April 2, 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 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. DPR-53 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 310, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 75 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: July 16, 2015

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-318

Amendment No. 288
Renewed License No. DPR-69

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated September 18, 2014, as supplemented by letters dated February 17, 2015, and April 2, 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 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. DPR-69 is hereby amended to read as follows:

Enclosure 2

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 288, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 75 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael I. Dudek, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 16, 2015

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 310 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 288 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	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 Pages	Insert Pages
5.5-18	5.5-18
5.5-19	5.5-19
5.5-20	5.5-20

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and 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) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state reactor core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 310, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 227 to Facility Operating License No. DPR-53, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 227. For SRs that existed prior to Amendment 227, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 227.

(3) Additional Conditions

The Additional Conditions contained in Appendix C as revised through Amendment No. 305 are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Additional Conditions.

(4) Secondary Water Chemistry Monitoring Program

Exelon Generation shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and 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) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now and hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at reactor steady-state core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 288 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 201 to Facility Operating License No. DPR-69, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 201. For SRs that existed prior to Amendment 201, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 201.

(3) Less Than Four Pump Operation

The licensee shall not operate the reactor at power levels in excess of five (5) percent of rated thermal power with less than four (4) reactor coolant pumps in operation. This condition shall remain in effect until the licensee has submitted safety analyses for less than four pump operation, and approval for such operation has been granted by the Commission by amendment of this license.

(4) Environmental Monitoring Program

If harmful effects or evidence of irreversible damage are detected by the biological monitoring program, hydrological monitoring program, and the

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A dated October 2008.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.7 psig. The containment design pressure is 50 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.16 percent of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.
   2. For each door, leakage rate is $\leq 0.0002\ L_a$ when pressurized to $\geq$ 15 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of CRE and the CRE boundary.

b. Requirements for maintaining CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. License controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the 36 month assessments of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by

5.5 Programs and Manuals

the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and assessing the CRE boundary as required by paragraphs c and d respectively.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 310 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 288 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By application dated September 18, 2014 (Reference 1), as supplemented by letters dated February 17, 2015 (Reference 2), and April 2, 2015 (Reference 3), Exelon Generation Company, LLC (Exelon, the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (CCNPP), Technical Specifications (TSs). The supplemental letters dated February 17, 2015, and April 2, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on November 25, 2014 (79 FR 70214).

The proposed changes would revise the CCNPP TS 5.5.16, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163 (September 1995) (Reference 4), with a reference to Topical Report (TR) [Nuclear Energy Institute] NEI 94-01, Revision 3-A (Reference 5), and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC safety evaluation report (SER) in NEI 94-01, Revision 2-A, (Reference 6) dated October 2008. These references are the implementing documents for the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B, performance-based primary containment leakage testing program for CCNPP. The proposed change would allow an increase in the Type A test interval from the current 10 years to a maximum of 15 years and allow an increase in the Type C test interval from the current 60 months to 75 months. The proposed change would also delete the one time exceptions granted to the Type A test interval and exceptions from the post-modification Type A test when the steam generators at CCNPP were replaced.

2.0 REGULATORY EVALUATION

The construction permits for CCNPP were issued by the Atomic Energy Commission (AEC) on July 7, 1969 (Reference 7), and the operating licenses were issued on July 31, 1974, for Unit No.1 (Reference 8) and August 13, 1976 for Unit No.2 (Reference 9). The AEC published the final rule that added 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule becoming effective on May 21, 1971. As stated in SECY-92-223, dated September 18, 1992 (Reference 10), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. The CCNPP updated final safety analysis report (UFSAR), Revision 47, dated August 27, 2014 (Reference 11), states that the plant was designed and constructed to meet the intent of the GDC published in July 1967. The plant's GDC are discussed in the UFSAR, Appendix 1C, "AEC Proposed General Design Criteria for Nuclear Power Plants."

2.1 Component Description

As stated in the CCNPP UFSAR Section 5.1.2,

> The [containment] structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab ... The entire interior surface of the structure was lined with a 1/4" thick welded American Society for Testing and Materials (ASTM) A36 [carbon] steel plate ... The liner plate was protected from corrosion on the inside with 3 mils of inorganic zinc primer topped with 6 mils of an organic epoxy up to Elevation 75'0", and 3 mils of an inorganic topcoat above that elevation. There is no paint on the side [that comes] in contact with concrete.

Several systems penetrate the containment structure wall through welded steel penetrations. The steel liner and its penetrations establish the leakage-limiting boundary of the containment structure.

The design function of the containment structure is to contain all radioactive material released from the reactor core following a postulated design basis Loss-of-Coolant Accident (LOCA). The structure serves as both a biological shield and a pressure container. The containment internal design pressure is 50 psig, with a concrete design surface temperature of 276°F and a design leak rate of 0.16-percent by weight per day at design pressure and temperature.

2.2 Regulatory Requirements

The regulations in 10 CFR 50.54(o) require that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50 (Reference 17), "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 does not exceed allowable leakage rate values specified in the TSs or associated bases and that 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, the Type A, B, and C 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 the 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 TSs. 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 Part 50, Appendix J, Option B, requires that the RG or other implementation document used by a licensee to develop a performance-based leakage-testing program be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain a justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a RG.

The implementation document that is currently referenced in the CCNPP TS 5.5.16, "Containment Leakage Rate Testing Program," is RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 4). In RG 1.163, TR NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," dated July 26, 1995, is endorsed as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50, subject to four regulatory positions delineated in Section C of the RG. In NEI 94-01, Revision 0, provisions are included that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

In TR NEI 94-01, Revision 2-A, an approach for implementing the optional performance-based requirements of Option B in Appendix J to 10 CFR Part 50, is described. It incorporates the regulatory positions stated in RG 1.163 (September 1995), and includes provisions for extending Type A test intervals to up to 15 years. In the related NRC SER dated June 25, 2008 (Reference 13), the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B in Appendix J to 10 CFR Part 50, 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 SER. Section 3.1 of this SER 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 inservice inspection (CISI) requirements mandated by 10 CFR 50.55a to the containment leak rate testing requirement.

Guidance for extending Type C Local Leak Rate Test (LLRT) intervals beyond 60 months is given in TR NEI 94-01, Revision 3-A (Reference 5). This report states that:

> Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If the Type B and C test results are not acceptable, the test frequency should be set at the initial test intervals. Once the cause determination and corrective actions have been completed, acceptable performance may be reestablished and the testing frequency returned to the extended intervals as specified in this document.

As described in NRC letter 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,'" dated August 20, 2013 (Reference 17), Revision 3-A inadvertently did not include the limitations and conditions provided in NRC's June 25, 2008, SER approving NEI 94-01, Revision 2. The letter states that "Any licensee submissions referencing the TR [Revision 3-A] will require requests for additional information from the NRC to address the limitations and conditions from the NRC [SER] for NEI 94-01, Revision 2."

The regulations at 10 CFR 50.36(c)(5), "Technical Specifications," require, in part, the inclusion of administrative controls in TSs that are necessary to ensure operation of the facility in a safe manner. This license amendment request requests a change to a TS under the "Administrative Controls" section of the CCNPP TSs.

The regulations at 10 CFR 50.55a, "Codes and Standards," contains the containment inservice inspection requirements that, in conjunction with the requirements of Appendix J, ensure the continued leak tightness and structural integrity of the containment during its service life.

The regulations at 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," states, in part, that the licensee shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions.

The licensee cited the following license amendments approved by the NRC that extended the Type A test interval from 10 years to 15 years. Consequently, the NRC staff reviewed the SEs (safety evaluations) associated with these license amendments:

- Nine Mile Point Nuclear Station, Unit 2 – Agencywide Documents Access and Management System (ADAMS) Accession Number ML100730032 (Reference 14)

- Arkansas Nuclear One, Unit 2 – ADAMS Accession Number ML110800034 (Reference 15)
- Palisades Nuclear Plant – ADAMS Accession Number ML120740081 (Reference 16)

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

The licensee proposes to change the CCNPP TSs so as to allow it to extend the intervals for Type A tests to 15 years and for Type C tests to 75 months.

The CCNPP TS 5.5.16, "Containment Leakage Rate Testing Program," currently states, in part, that:

> A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:
>
> a. Nuclear Energy Institute (NEI) 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. The first Unit 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016.
>
> b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement
>
> c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

In its license amendment request (LAR) dated September 18, 2014, the licensee proposed to remove from TS 5.5.16 exceptions (a), (b), and (c), and replace the reference to RG 1.163 (September 1995) with a reference to TR NEI 94-01, Revision 3-A. The proposed change would revise TS 5.5.16 to state, in part, that:

> A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. 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.

The NEI 94-01, Revisions 2-A (Reference 6) and 3-A (Reference 5), have been approved by the NRC. These TRs define a performance-based approach for determining Type A, Type B, and Type C containment leakage rate testing frequencies and provide methods for justifying the extension of these test intervals based on performance history and risk insights. Revision 2-A provides an acceptable approach for implementing the performance-based requirements of Option B in Appendix J to 10 CFR Part 50, subject to the limitations and conditions noted in Section 4.0 of the associated NRC SER. It provides guidance for extending the Type A test interval up to 15 years and incorporates the regulatory positions stated in RG 1.163 (September 1995).

Revision 3-A, in addition to guidance for the Type A test interval, includes guidance for extending the Type C test interval up to 75 months. However, Revision 3-A omitted the limitations and conditions for the Type A test that were required by the NRC in Section 4.0 of its SER for Revision 2-A. The NRC notified NEI by a letter dated August 20, 2013 (Reference 17) that, due to this omission, Revision 3-A will not be referenced in the update to RG 1.163, and that any licensee submission referencing Revision 3-A will require Requests for Additional Information (RAIs) from the NRC to address these limitations and conditions.

Accordingly, by RAI, the NRC staff informed the licensee that Revision 3-A of NEI TR 94-01 did not include the limitations and conditions contained in Revision 2-A. Therefore, the NRC staff requested the licensee to address the limitations and conditions documented in Section 4.0 of the NRC staff's SER for NEI TR 94-01, Revision 2-A.

In response to this RAI (Reference 2), the licensee agreed to address the limitations and conditions specified in NEI 94-01, Revision 2-A, in its TS 5.5.16 and, thus, proposed to revise the TS 5.5.16 to state, in part, that:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

3.2 Deterministic Considerations: Structural and Leak-Tight Integrity of the Containment

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

As described in the LAR (Reference 1, Section 3.1) and Section 5.1 of the CCNPP UFSAR, the containment structure consists of a reinforced pre-stressed concrete cylinder and dome supported by a flat foundation slab. The circular cylinder wall is pre-stressed by a system of horizontal and vertical tendons. The horizontal tendons are anchored at buttresses equally spaced around the outside of the containment and the vertical tendons are anchored to the base slab at the bottom and to the ring girder at the top. The dome has a 3-way post-tensioning system. The foundation slab is conventionally reinforced with reinforcing steel. A welded steel liner is attached to the inside face of the containment structure to assure leak tightness. A finished concrete floor covers the portion of the liner on the containment foundation slab. A leak chase system allows the containment liner welds located under the concrete floor to be leak tested during the Integrated Leak Rate Test (ILRT) of the containment.

The CCNPP Units 1 and 2 leak-tight integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity and structural integrity of the containment is verified through a Type A ILRT, as required by Appendix J to 10 CFR Part 50. The leakage rate testing requirements of Option B in Appendix J to 10 CFR Part 50 (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.

3.2.2 Historical Plant-Specific Containment Leakage Testing Program Results

As indicated in the LAR and the current CCNPP TS, the maximum allowable containment leakage rate, L_a, is 0.16 percent of containment air weight per day at the peak calculated containment internal pressure for the design basis LOCA, P_a.

In Section 3.2.4 of the LAR, the licensee provided the history of the CCNPP Type A test results. The leakage rate for the last two Type A tests for CCNPP Unit 1, performed on July 5, 1992, and May 3, 2006, were 0.1564 and 0.0952 weight percent per day, and for CCNPP Unit 2, performed on May 2, 2001, and March 17, 2013, were 0.0738 and 0.0802 weight percent per day; all of which are below the current maximum allowable containment leakage rate of 0.16 weight percent per day.

In Section 3.4.4 of the LAR, the licensee stated that the CCNPP Type B and Type C testing program requires testing of electrical penetrations, airlock hatches, flanges, and valves within the scope of the program as required by Option B in Appendix J to 10 CFR Part 50, and CCNPP TS 5.5.16.

The licensee provided the combined Type B and Type C LLRT leak rate from 2008 through 2014 for CCNPP Unit 1, and from 2003 through 2013 for CCNPP Unit 2 indicating a considerable margin between the leakage rate values and the TS 5.5.16 limit of 0.6 L_a.

The licensee stated that industry experience has shown that the Type B and C tests can identify the vast majority of all potential containment leakage paths, and that Type B and Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained.

Based on a review of the licensee's LAR and responses to RAIs, and based on the information discussed above, the NRC staff concludes that: (1) the performance history of successful completion of the two most recent consecutive periodic 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 well below the acceptance criterion; and (3) there is reasonable assurance that the licensee is effectively implementing its Type B and Type C testing program under Option B in Appendix J to 10 CFR Part 50. These conclusions support approving the extension of the Type A test interval at CCNPP from 10 to 15 years.

3.2.3 Containment In-Service Inspection (CISI) Program

In Section 3.4.2 of the LAR, the licensee stated that the CCNPP Units 1 and 2 CISI program is established in accordance with 10 CFR 50.55a. This program has been developed to comply with American Society of Mechanical Engineers (ASME) Code, Section XI, 2004 Edition, except where specific written alternatives from the ASME Code requirements have been requested and granted by the NRC. General visual examinations of accessible interior and exterior surfaces of the containment structure are conducted in accordance with the CCNPP 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).

As noted in Section 3.3.2 of this SE, in addition to the inspections performed in accordance with the CISI program, supplemental inspections in accordance with approved plant surveillance test procedure (STP)-M-665-1 and STP-M-665-2 are utilized to perform visual inspection of the accessible internal and exterior surfaces of the containment structure.

The IWE/IWL inspections and supplemental inspections, in accordance with other approved plant procedures, are used to satisfy the general visual examination requirements of Option B in Appendix J to 10 CFR Part 50, and to monitor and manage the age-related degradations of the containment to ensure that containment structural and leak-tight integrity is maintained through its service life.

The CCNPP operating experience and corrective actions taken to address the degradation of the moisture barrier and liner corrosion, containment concrete surface degradation, and vertical tendons corrosion have been discussed in Section 3.3.3 of this SE and will not be repeated here.

In Section 3.4.1 of the LAR, the licensee stated that CCNPP conducts condition assessments of Service Level I coatings inside containment as part of the safety-related and controlled protective coatings program. The inspections of coating systems are scheduled every outage on a pre-established basis to verify containment liner coating thickness and condition. These inspections provide another opportunity to identify potential degradation of the containment liner surfaces.

In its letter dated April 2, 2015 (Reference 3), the licensee's response to the NRC staff's RAI regarding the NRC Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner" (Reference 18), stated that: (1) the CCNPP design configuration is different from the one considered as "typical" in Information Notice 2014-07; (2) at CCNPP, the pipe protruding from the leak chase channel is either "flush" to the floor or exposed to the surface with no access box and cover plate, which configuration eliminates the chances of moisture accumulation thus avoiding degradation of this area; and (3) all visual inspections performed on the accessible surfaces of the leak chase channel system have been satisfactory with no degradation identified.

Furthermore, as noted in Section 3.6 of the LAR, the CCNPP containment structures are in scope for license renewal based on 10 CFR 54.4(a). Table 16-2 of the CCNPP UFSAR lists the aging management programs that are associated with containment system and components along with the aging mechanism and the aging management program.

Based on the above, the NRC staff finds that the licensee has adequate in-service inspection programs in place to monitor and manage age-related degradation of the CCNPP containment structures. Also, the CCNPP operating experience and the results of the inspections, to date, indicate that the structural and leak-tight integrity of the containment have been effectively monitored and managed and will continue to be effectively managed if the ILRT interval is extended from 10 years to 15 years. These conclusions support approving the extension of the Type A test interval at CCNPP from 10 to 15 years.

3.2.4 Containment Leak-Tight Integrity Tests

The overall leak-tight integrity and structural integrity of the containment is verified through 10 CFR Part 50, Appendix J, Type A tests. The leak-tight integrity of the containment penetrations such as equipment hatch, airlocks, flanges, and electrical penetration is verified through 10 CFR Part 50, Appendix J, Type B tests. The leak-tight integrity of the containment isolation valves is verified through 10 CFR Part 50, Appendix J, Type C tests. These tests are performed at the calculated peak containment internal pressure related to the design basis LOCA. The leakage rate testing requirements of the Type A, B, and C tests along with the CISI requirements mandated by 10 CFR 50.55a help ensure the continued leak-tight and structural integrity of the containment during its service life.

3.2.5 Current Leak Rate Test Frequency Requirements

The current Type A test interval is 10 years and Type C test interval is 60 months.

The licensee stated that the current TS 5.5.16, "Containment Leakage Rate Testing Program," requires compliance with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. According to TS 5.5.16, the testing is required to be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 4), including errata, as modified by the following exceptions:

(a) Nuclear Energy Institute (NEI) 94-01 - 1995, Section 9.2.3 [Reference 19]: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. The first Unit 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016.

(b) Unit 1 is excepted from post modification integrated leakage rate testing requirements associated with steam generator replacement.

(c) Unit 2 is excepted from post modification integrated leakage rate testing requirements associated with steam generator replacement.

In RG 1.163, Regulatory Position C.1 (Reference 4), it states that:

> ...licensees intending to comply with Option B in the amendment to Appendix J [to 10 CFR Part 50] should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 [Revision 0], rather than using the test intervals specified in [American National Standards Institute] ANSI/[American Nuclear Society]ANS-56.8-1994. All other technical methods and techniques for performing Types A, B, and C tests contained in ANSI/ANS-56.8-1994 are acceptable to the NRC staff.

In NEI 94-01, Revision 0, Section 11.0 (Reference 19), it provides guidance on establishing leakage testing frequencies of Type A, B, and C tests and assessing the risk impact using historical performance data.

In NEI 94-01, Revision 0, Section 9.2.3 (Reference 19), it states that:

> Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a [where L_a is the maximum allowable leakage rate at the calculated peak containment internal pressure, P_a, related to the design basis accident specified in the TS] ... Elapsed time between the first and the last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

In NEI 94-01, Revision 0, Section 10.1, it states that Type C testing shall be performed:

> Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for the recommended surveillance frequency for Type B and Type C testing given in this section may be extended by up to 25 percent of the test interval, not to exceed 15 months.

In NEI 94-01, Revision 0, Section 10.2.3.2, it states that:

> Test intervals for Type C valves may be increased based upon completion of two consecutive periodic As-found Type C tests where the result of each test is within a licensee's allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the valve prior to implementing Option B to Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 24 months up to a maximum of 120 months.

3.2.6 Type A Test History

By letter dated August 29, 2007 (Reference 20), the CCNPP TS 5.5.16 L_a was changed from 0.2 percent to 0.16 percent of containment air weight per day at P_a, which is 49.7 psig. In its LAR, the licensee provided the leakage rate results of Type A tests performed during the last 7 outages of each unit. All results for the tests performed before and after August 29, 2007, meet the previous and the current TS 5.5.16 leakage requirement. The results demonstrate that the containment structure for both units is a leak-tight barrier.

3.2.7 Type B and C Test History

The CCNPP TS 5.5.16 requirement for the allowable maximum pathway total Types B and C leakage is 0.6 L_a. In the LAR, the licensee provided the following average and high values of the as-found and as-left combined Type B and C leak rates extracted from the results of tests performed from the spring of 2008 through the spring of 2014 for Unit 1 and from the spring of 2003 through the spring of 2013 for Unit 2.

- The as-found minimum pathway leak rate average for Calvert Cliffs Unit 1 shows an average of 6.1% of 0.6 L_a with a high of 8.5% or 0.051 L_a.
- The as-left maximum pathway leak rate average for Calvert Cliffs Unit 1 shows an average of 6.9% of 0.6 L_a with a high of 9.9% or 0.060 L_a.
- The as-found minimum pathway leak rate average for Calvert Cliffs Unit 2 shows an average of 8.2% of 0.6 L_a with a high of 12.4% or 0.074 L_a.
- The as-left maximum pathway leak rate average for Calvert Cliffs Unit 2 shows an average of 7.2% of 0.6 L_a with a high of 9.4% or 0.057 L_a.

The results show a history of successful tests because no as-found failure resulted in exceeding the TS 5.5.16 limit of 0.6 L_a. The licensee stated that the as-found minimum pathway summations represent the high quality of maintenance of the tested components while the as-left maximum pathway summations represent effective management of the containment leakage rate testing program. The staff reviewed the summary of results and agrees that the test results ensure that containment integrity is maintained because there is significant margin between the actual as-found outage summations and the performance criterion identified in the CCNPP TSs. These conclusions support approving the extension of the Type A test interval at CCNPP from 10 to 15 years.

3.2.8 Deterministic Evaluation Summary

Based on the NRC staff's review of the regulatory and technical evaluations in the licensee's LAR dated September 18, 2014 (Reference 1), as supplemented by letters dated February 17, 2015 (Reference 2), and April 2, 2015 (Reference 3), the NRC staff concludes that the licensee has effectively implemented an adequate containment leakage rate testing (i.e., ILRT and LLRT) program, CISI program, and supplemental inspections to periodically examine, monitor, and manage age-related degradation of the CCNPP containment structures. The results of the past ILRTs and LLRTs and the CISI program demonstrate acceptable performance of the CCNPP containment structures and demonstrate that the structural and leak-tight integrity of the containment structure is adequately managed. The NRC staff also finds that the structural and leak-tight integrity of the CCNPP containment structures will continue to be periodically monitored and effectively managed by the LLRT and CISI programs, if the current ILRT interval is extended from 10 years to 15 years. 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 ILRT interval at CCNPP is extended to 15 years.

3.3 NRC Staff Assessment of Licensee's Responses to Limitations and Conditions in NEI 94-01 Revision 2-A for Type A Test

As described in NRC letter dated August 20, 2013 (Reference 17), 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,'" NEI 94-01, Revision 3-A, inadvertently did not include the six limitations and conditions in NRC's June 25, 2008, SER approving NEI 94-01, Revision 2 (Reference 13). Although the six limitations and conditions were not included in NEI 94-01, Revision 3-A (Reference 5), they apply to a licensee's request to use NEI 94 01, Revision 3-A, to extend the ILRT interval. Therefore, as provided below, the NRC staff evaluated whether the licensee had adequately addressed and satisfied these six limitations and conditions as presented in its LAR, Table 3.7-1, and its supplemental letters.

3.3.1 Limitation/Condition 1:

The NRC Limitation/Condition 1 states:

> 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 3.7 of the LAR that CCNPP 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 complying 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 NRC Limitation/Condition 1.

3.3.2 Limitation/Condition 2:

The NRC Limitation/Condition 2 states:

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

The licensee provided a schedule of containment inspections in Sections 3.4.2 and 3.4.3 of the LAR. In NEI 94-01, Section 9.2.3.2, "Supplemental Inspection Requirements," it 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.

In the LAR, the licensee stated that: (1) the purpose of the CCNPP CISI program is to periodically perform examination of ASME Class Metallic Containment and Concrete Containment components in order to identify the presence of any service-related degradation; (2) the CISI program is established in accordance with 10 CFR 50.55a, such that the program has been developed to comply with ASME Boiler and Pressure Vessel (BPV) Code , Section XI, 2004 Edition, except where specific written alternatives from ASME BPV Code requirements have been requested and granted by the NRC; (3) the second 10-year CISI interval for CCNPP for the performance of IWE and IWL examinations began on September 9, 2009 and will end on September 9, 2018; (4) the third 10-year CISI interval will be effective between September 9, 2018 and September 9, 2028; (5) the IWL inspection period is five years, with two periods per inspection interval; and (6) the IWE inspection interval is divided in three periods.

In Section 3.4.3, "Supplemental Inspection Requirements," of the LAR, the licensee stated that in addition to the inspections performed in accordance with the CISI program, STP-M-665-1 and STP-M-665-2, "Containment Visual Inspection," are utilized to perform visual inspections of the accessible internal and exterior surfaces of the containment structure to identify evidence of structural deterioration, which could affect either structural integrity or leak tightness. The licensee indicated that in accordance with STP-M-665-1 and STP-M-665-2, the containment liner is inspected during each refueling outage and prior to each Type A test, and the containment concrete is inspected every 36 months (plus or minus 14 months) and prior to every Type A test.

The licensee concluded that, together, these examinations assure that at least four general visual examinations of containment concrete and metallic surfaces will be conducted before the next Type A test if the Type A test interval is extended to 15 years and; therefore, the requirements of Section 9.2.3.2 of NEI 94-01 and NRC Limitation/Condition 2 are met.

On the basis that the CCNPP IWE and IWL inspections, and the supplemental inspections in accordance with STP-M-665-1 and STP-M-665-2, as described above, results in at least three general visual 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, therefore, satisfies NRC Limitation/Condition 2.

3.3.3 Limitation/Condition 3:

The NRC Limitation/Condition 3 states:

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

The licensee stated that procedures STP-M-665-1 and STP-M-665-2, "Containment Visual Inspection," are utilized to perform visual inspections of the normally accessible internal and exterior surfaces of the primary containment to identify evidence of structural deterioration, which could affect either structural integrity or leak tightness. These are scheduled surveillance tests and are performed during each refueling outage. In Section 3.5 of the LAR, the licensee provided the following operating experience and indications of degradation that have been identified, placed into the CCNPP corrective action programs, and corrected.

3.3.3.1 Moisture Barrier Seal Degradation and Liner Corrosion

In Section 3.5.1 of the LAR, the licensee stated that: (1) inspections on the containment liner are currently conducted in accordance with Examination Category E-A of the ASME Code, Section XI, 2004 Edition, and are performed such that 100 percent of the accessible portion of the liner is inspected during each inspection period; (2) the portion of the liner that covers the containment foundation slab is considered inaccessible and its acceptability must therefore be evaluated whenever conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to the inaccessible area; and (3) the moisture barrier seal, located at the interface of the containment floor slab and the containment wall liner, is examined so that 100 percent of the seal is visually examined during each inspection period.

The licensee further stated that CCNPP discovered significant age related degradation of the Unit 1 moisture barrier seal in 1994. As part of the corrective actions, a decision was made to subsequently replace the moisture barrier seal for both Units 1 and 2. In 1999, during the replacement of the Unit 2 moisture barrier seal, areas of pitting and general corrosion of the liner were discovered at the interface between the floor slab and the containment wall liner under the moisture barrier seal. The evaluation of the liner condition concluded that it was acceptable to return the liner to service without repair of the degraded area.

The replacements of the Unit 1 and 2 moisture barrier seals have been completed. The most recent inspections of the containment liner and moisture barrier seal indicate that the replacement of the moisture barrier seal has arrested the corrosion and pitting throughout the affected area and has prevented any new areas of corrosion and pitting from occurring.

The Unit 2 moisture barrier seal continues to be subject to augmented inspections due to the identification of a crack and subsequent repair of the seal during the 2013 refueling outage. In response to the NRC staff's RAI, the licensee, in a letter dated February 17, 2015 (Reference 2), stated that a 1 inch crack was identified on the moisture barrier near a leak chase channel. It was determined that there was no liner impact due to moisture or evidence of moisture intrusion. The crack was repaired and a subsequent examination is scheduled for performance during the 2015 Unit 2 refueling outage.

3.3.3.2 Containment Concrete Surface Degradation

In Section 3.5.2 of the LAR, the licensee stated that the reinforced concrete portions of the containment structure are inspected in accordance with Examination Category L-A of the

ASME Code, Section XI, 2004 Edition. A 100 percent visual examination of CCNPP containment concrete surfaces is conducted every five years.

The licensee further stated that during the 2005 and 2007 inspections of CCNPP containment concrete surfaces, grease leaks, efflorescence, and other stains were identified and entered into the CCNPP corrective action program for resolution. In response to the NRC staff's RAI, the licensee stated, in letters dated February 17, 2015 (Reference 2), and April 2, 2015 (Reference 3), that: (1) the evaluation of the identified tendon grease leakage categorized the findings as of 'low safety significance' and characterized the leakage as grease cap leaks requiring cleaning and/or joint sealant replacement; and (2) the two commitments made in Attachment 4 of the LAR to complete concrete repairs to areas of the containment structure have been completed. The repairs have addressed all issues previously identified with containment dome weathering and the effects of freeze/thaw cycles. In addition, repairs associated with the concrete delamination around the sloped surface above the equipment hatch have been completed. All corrective actions associated with these issues, including removal of loosened concrete, cleaning of grease leaks, and sealant application were completed in December 2014.

3.3.3.3 Vertical Pre-stressing Tendon Corrosion

In Section 3.5.3 of the LAR, the licensee stated that during the 20-year Unit 1 tendon surveillance in 1997, several vertical tendons selected for the lift-off testing were found to have broken and corroded wires at the top ends of the tendons just below the stressing washer. The discovery of broken wires in these tendons initiated an expansion of the Unit 1 vertical tendon inspection scope to perform visual inspections and lift-off testing on all Unit 1 vertical tendons. Subsequently, broken and corroded wires were found throughout the Unit 1 vertical tendon population at the top ends of the tendons. Following completion of the Unit 1 surveillance, the 20-year surveillance of the Unit 2 tendons was conducted and abnormal conditions very similar to Unit 1 were found on the Unit 2 vertical tendons.

As part of the corrective action plan, all of the vertical tendons were re-greased using a new corrosion inhibiting grease. In addition, 46 vertical tendons in each unit were replaced in 2001 and 2002, and had new grease put in place at that time. At the end of these corrective actions, all of the vertical tendons had a redesigned pressure-tight, grease-filled cap installed at the upper-bearing plate to prevent water intrusion. The bottom grease cap for every vertical tendon was also replaced with a new redesigned pressure-tight grease cap.

The licensee, in addition to the regular inspections of tendons according to ASME Code, Section XI, conducted two additional inspections of the anchorhead and buttonhead region of a selected sample of vertical tendons in 2005 and 2007. No new issues were identified as a result of these inspections. Furthermore, the licensee stated that the evaluation of the in-service inspection results for the 35th year, conducted in 2012 for Unit 1 and in 2013 for Unit 2 containment structures, concluded that no abnormal degradation of the post-tensioning systems has been experienced.

3.3.3.4 Inaccessible Areas

In Section 3.4.2 of the LAR, the licensee stated that the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas is evaluated for CCNPP. Also, the licensee stated that, to date, CCNPP has not needed to implement any new technologies to perform inspections of any inaccessible areas. However, as noted in the LAR, the licensee actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. The licensee also continuously reviews the industry operating experience to determine its applicability to CCNPP. As also noted in the LAR, the licensee would explore adjustments to inspection plans and availability of new, commercially available, technologies for the examination of the inaccessible areas of the containment as part of these activities.

Based on the above information provided by the licensee regarding the CCNPP operating experience and corrective actions taken to address the degradation of the moisture barrier and liner corrosion, containment concrete surface degradation, and vertical tendons corrosion; as well as based on the fact that the licensee, to date, had not identified any conditions that would indicate the presence of any potential degraded conditions in inaccessible areas of the concrete containment structure and steel liner, the NRC staff concludes that the licensee has satisfied NRC Limitation/Condition 3.

3.3.4 Limitation/Condition 4:

The NRC Limitation/Condition 4 states:

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

The licensee stated in Section 3.7 of its LAR that the CCNPP steam generators were replaced in 2002 and 2003, respectively. Furthermore, the licensee stated in its letter dated February 17, 2015 (Reference 2), that there are no planned modifications for CCNPP that will require a Type A test prior to the next Units 1 and 2 Type A tests proposed under this LAR. There is also no anticipated addition or removal of plant hardware within the containment building, which could affect its leak-tightness.

Based on the above, the NRC staff concludes that the licensee's program will implement the staff's 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 staff's SER for NEI 94-01, Revision 2. Therefore, the NRC staff concludes that the licensee has satisfied NRC Limitation/Condition 4.

3.3.5 Limitation/Condition 5:

The NRC Limitation/Condition 5 states:

> 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 3.7 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 (Reference 21).

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 extension requests. Therefore, the NRC staff concludes that the licensee has satisfied NRC Limitation/Condition 5.

3.3.6 Limitation/Condition 6:

The NRC Limitation/Condition 6 states:

> 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 94-01, Revision 2, and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, including the use of past containment ILRT data.

The licensee stated in its LAR that this condition is not applicable to CCNPP because CCNPP are not licensed under 10 CFR Part 52. The NRC staff concludes that Condition 6 does not apply to CCNPP because CCNPP is currently an operating reactor licensed under 10 CFR Part 50.

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

3.4 Staff Assessment of Licensee's Responses to Limitations and Conditions in NEI 94-01 Revision 3-A for Type C Tests

In Section 4.0, "Limitations and Conditions," of the NRC staff SER for TR NEI 94-01, Revision 3-A (Reference 5), the NRC staff identified two conditions for using this TR for Type C tests. As per Section 3.2 of the SER for TR NEI 94-01, Revision 3-A (Reference 5), the failure of any

Type C tests would reset the testing interval back to 30 months until the acceptable performance history of the last two as-found component tests is re-established. Also, the extension will not apply to valves that are restricted to 30-month test intervals, or to valves held to 30-month intervals due to unacceptable performance.

These conditions and the NRC staff assessment of the licensee's responses presented in the LAR are as follows.

3.4.1 Condition 1

Condition 1 states:

> NEI TR 94-01, Revision 3-A, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI TR 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

Licensee's Response to Condition 1

The licensee stated in the LAR that:

> The post-outage report shall include the margin between the Type B and Type C Minimum Pathway Leak Rate (MNPLR) summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of $0.60L_a$.
>
> When the potential leakage understatement adjusted Type B & C MNPLR total is greater than the Calvert Cliffs administrative leakage summation limit of $0.50L_a$, but less than the regulatory limit of $0.6L_a$, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the Calvert Cliffs administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

> Calvert Cliffs will apply the 9-month grace period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

NRC Staff Assessment

For allowing the Type C test interval extension from 60 to 75 months, the NRC staff finds the responses to Condition 1 acceptable because the licensee agrees to: (1) adequately address the post-outage test report requirement, (2) necessary corrective actions when the adjusted Type B and C MNPLR total is greater than the administrative leakage summation limit of 0.50 L_a set by the licensee which is conservatively less than the regulatory limit of 0.6 L_a, and (3) extend the interval from 75 months to 84 months for eligible components under non-routine emergent conditions only.

3.4.2 Condition 2

Condition 2 states:

> The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1. When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing

program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Licensee's Response to Condition 2

The licensee stated in the LAR that:

> The change in going from a 60 month extended test interval for Type C tested components to a 75 month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25 percent in the LLRT periodicity. As such, Calvert Cliffs will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the As-Left leakage total for each Type C component currently on the 75 month extended test interval. This will result in a combined conservative Type C total for all 75 month LLRT being "carried forward" and will be included whenever the total leakage summation is required to be updated (either while on line or following an outage). When the potential leakage understatement adjusted leak rate total for those Type C components being tested on a 75 month extended interval is summed with the non-adjusted total of those Type C components being tested at less than the 75 month interval and the total of the Type B tested components, if the Minimum pathway leakage rate is greater than the Calvert Cliffs administrative leakage summation limit of $0.50L_a$, but less than the regulatory limit of $0.6L_a$, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the Calvert Cliffs administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.
>
> If the potential leakage understatement adjusted leak rate Minimum pathway leakage rate is less than the Calvert Cliffs administrative leakage summation limit of 0.50 La, then the acceptability of the 75 month LLRT extension for all affected Type C components has been adequately demonstrated and that the calculated local leak rate total represents the actual leakage potential of the penetrations.
>
> In addition to Condition 1, Parts 1, 2 which deal with the MNPLR Type B & C summation margin, NEI 94-01, Revision 3-A also has a margin related requirement as contained in Section 12.1, Report Requirements:
>
> A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a

record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

At Calvert Cliffs in the event an adverse trend in the aforementioned potential leakage understatement adjusted Type B & C summation is identified, then an analysis and determination of a corrective action plan shall be prepared to restore the trend and associated margin to an acceptable level. The corrective action plan shall focus on those components which have contributed the most to the adverse trend in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

At Calvert Cliffs an adverse trend is defined as three (3) consecutive increases in the final pre-RCS Mode change Type B & C MNPLR leakage summation value, as adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of L_a.

NRC Staff Assessment

For increasing the Type C test interval from 60 to 75 months, the NRC staff finds the licensee's responses to Condition 2 acceptable based on the following:

1. For the purposes of monitoring or trending the overall containment leakage potential, the licensee proposes to estimate the potential understatement by conservatively applying an adjustment factor of 1.25 to the as-left leakage total for each Type C component on the 75 month extended test interval. The licensee's rationale for the adjustment factor is that the change in the test interval from 60 months to 75 months represents a 1.25 times increase and is reasonable. Therefore, the as-found minimum pathway leakage rates for the just tested penetrations will be summed up with the 1.25 times as-left minimum pathway leakage rates for penetrations tested during the previous 1, 2, or even 3 refueling outages.

2. The leakage rate testing program trending will include an estimate of the amount of understatement in the Type B and C total in a licensee's post-outage report. The report will provide reasons for the acceptability of the test interval extension, and demonstrate that the totals calculated represent the actual leakage potential of the penetrations. In case an adverse trend in the potential leakage understatement adjusted Type B and C summation is identified, the licensee will initiate a corrective action to restore the trend and associated margin to an acceptable level.

The corrective action plan will mention those components that have contributed the most to the adverse trend in the leakage summation value. The plan will provide the method of timely and appropriate correction that will prevent future leakage performance issues.

Based on the above evaluations, the NRC staff concludes that the licensee has adequately addressed and satisfied the two conditions in Section 4.1 of the NRC staff SER for NEI 94-01, Revision 3-A.

3.5 Probabilistic Risk Assessment (PRA)

3.5.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A (Reference 4), states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01, Revision 3-A, states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) TR 1018243,[1] "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (Reference 22). 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 its SER dated June 25, 2008 (Reference 13), the NRC staff found the methodology in NEI 94-01, Revision 2 (Reference 6), and EPRI TR-1009325, Revision 2 (Reference 22), acceptable for referencing by licensees proposing to amend their TS 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 (Reference 22), stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of 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.[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 3.2.4.5[3] of this SE [the SER for EPRI TR-1009325, Revision 2].

3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per

[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] 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.
[3] 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.

year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 La[4] [assigned a value of 100 times the maximum allowable La] instead of 35 La.

4. A LAR is required in instances where containment over-pressure is relied upon for [emergency core cooling system] ECCS performance.

3.5.2 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 3 of the LAR submitted September 18, 2014 (Reference 1), and was revised by the licensee in its letter dated February 17, 2015, in response to NRC RAIs (Reference 2).

In Section 3.3.1 of Attachment 1 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A[5] (Reference 5); the methodology described in EPRI TR-1009325, Revision 2-A (Reference 22); 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" (Reference 24). Additionally, the licensee used their methodology from the 2002 one-time ILRT extension license amendment for Unit 1 to assess the risk from undetected containment leaks due to steel liner corrosion. In response to an NRC RAI (Reference 2), the licensee reviewed the recent steel liner corrosion events and concluded that the corrosion sensitivity study from 2002 included in Section 5.1.4 of Attachment 3 to 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 related NRC SER. A summary of how each of these conditions has been met is provided in the sections below.

3.5.2.1 Technical Adequacy of the PRA

The first condition stipulates that the licensee submit 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 RIS 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 25), the NRC staff will use Revision 2 of RG 1.200 (Reference 23) to assess the 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 I of the ASME PRA

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[4] The term "La", instead of "L_a", is used in this context to be consistent with the quoted material.

[5] NEI 94-01, Revision 3-A (Reference 4), 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.

standard shall be applied as the standard for assessing PRA quality for IRLT extension applications, since approximate values of Core Damage Frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

Attachment 3 to the LAR states that the Calvert Cliffs Internal Events and Wind Model, Calvert-CAFTA-TREE-6.2a, was used to perform the plant-specific risk assessment. It also states that configuration control activities are routinely performed to ensure that the PRA model reflects the as-built, as-operated plant. The configuration control activities include reviewing design changes and procedure changes for their impact on the PRA model, reviewing any new engineering calculations and revisions to existing calculations for their impact on the PRA model and updating plant specific initiating event frequencies, failure rates, and maintenance unavailabilities based upon reviews of plant program data.

The licensee stated that the at-power, internal events PRA received a full scope peer review by the Pressurized Water Reactor Owners Group in June 2010, and was performed against the guidance of RG 1.200, Revision 2, and requirements of the ASME/American National Standards (ANS) RA-Sa-2009. The peer review found that 97 percent of the Supporting Requirements (SRs) were met with Capability Category II or better. Three SRs were considered "not met" and eight SRs were considered met at Capability Category I. Table 1 of Attachment 3 to the LAR lists the findings and observations (F&Os) identified in the 2010 peer review and evaluates their impact on the ILRT extension application. The licensee states that all findings which could be relevant to the application were updated in the internal events model. It further states that no follow-on or focused peer reviews were required because there were no significant changes and no new methods applied in the internal events PRA model. In response to an NRC RAI, the licensee identified the three SRs considered not met by the peer review team as LE-F2, LE-G5, and IFQU-A10. As reported by the licensee in Table 1 of Attachment 3 to the LAR, the findings related to SR LE-F2 and LE-G5 were addressed, and the finding related to SR IFQU-A10 is still open. The staff reviewed the disposition of the findings related to SR LE-F2 and, LE-G5 and agreed that they have been adequately addressed for this LAR. The licensee stated that IFQU-A10 relates to documentation of the treatment of the internal flood analysis and has no impact on the ILRT analysis. The staff agreed that the remaining open F&O IFQU-A10 has no impact on the LAR.

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.

In Section 5.3.1 of Attachment 3 to the LAR, the licensee performed an analysis of the impact of external events, which includes an evaluation of seismic and fire risks and a discussion of high winds risk. The licensee used the results from the Individual Plant Examination of External Events (IPEEE) to assess the seismic risk contribution. To assess the fire risk contribution, the licensee used the results from both the fire PRA submitted in support of the transition to the National Fire Protection Association (NFPA) 805 Standard, as well as the fire risk results from the IPEEE fire PRA analysis. In response to an NRC RAI (Reference 2), the licensee revised the external event analysis to correct the IPEEE CDF values for fire and seismic events. Reviewing the quality of detailed fire PRA models is beyond the scope of ILRT extension application reviews. The quality and results of the CCNPP fire PRA is currently under review as a part of the NRC review of the licensee's LAR to adopt NFPA 805. Therefore, the NRC staff relied on the fire risk analysis based on IPEEE results and finds those results sufficient to support the ILRT extension application.

The licensee also included in the LAR a discussion of high winds risk. Section 5.3.1 of Attachment 3 to the LAR states that the CCNPP topographical location presents the possibility of high wind events. These events include tornadoes, thunderstorms, freezing precipitation, and hurricanes. The LAR further states that these natural disasters are modeled in the internal events model. The licensee included in the LAR a discussion on the quality of the high winds PRA. The LAR states that:

> The Calvert Cliffs high winds PRA model is very conservative in the tornado area in that all tornados are grouped into the most conservative event. PRA risk for tornadoes and high winds are based upon IPEEE values. Calvert Cliffs has maintained and updated a high wind PRA model in order to perform risk assessment of tornado missile impacts and hurricane force winds. Although this model has not been peer reviewed in compliance with the ASME/ANS RA-Sa-2009 standard, the model is based upon accepted methodology and utilizes the ASME/ANS RA-Sa-2009 compliant internal events model. High winds updates are not expected to cause a significant increase in CDF or LERF.

The NRC staff finds the licensee's assessment of high winds acceptable for the ILRT extension application.

The licensee has also evaluated its PRA against the current ASME PRA standard and Revision 2 of RG 1.200, evaluated the findings for applicability to the ILRT interval extension, and addressed the findings or explained their impact. The NRC staff reviewed these findings and agrees that dispositioned findings have been adequately addressed for this application and the open finding from the peer reviews has no impact on the ILRT interval extension application. Furthermore, the licensee included a quantitative assessment of the contribution of external events. Therefore, as a result of these evaluations, 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.

3.5.3 Estimated Risk Increase

The second condition of EPRI TR-1009325, Revision 2 (Reference 22), stipulates that the licensee submit 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 (Reference 24) 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 (Reference 22). 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 per year or 1 percent 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 in Section 3.5.5 of this SE, CCNPP does not rely on containment over-pressure for ECCS performance. Thus, the associated risk metrics include LERF, population dose, and CCFP.

The licensee provided the results of the plant-specific risk assessment in Section 3.3.3 of Attachment 1 to the LAR. Details of the risk assessment are provided in Attachment 3 to the LAR, which was revised by the licensee in its letter dated February 17, 2015 (Reference 2), in response to NRC RAIs. The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR Part 50, Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF for internal events is 4.54E-08 per year for Unit 1 and 2.46E-08 per year for Unit 2. These values, reported in Section 3.3.3 of Attachment 1 to the LAR, do not include the steel liner corrosion effects. In Section 5.3.2 of Attachment 3 to the LAR, the licensee performed a sensitivity study on the steel liner corrosion likelihood. The licensee did not follow the guidance from EPRI TR-1009325, Revision 2-A, to perform this sensitivity study. Rather than individually modifying the various parameters that compose the final corrosion likelihood value, the licensee chose to directly multiply the corrosion likelihood by factors of 1,000, 10,000, and 100,000. The staff finds that the licensee's approach of multiplying the steel liner corrosion likelihood by a factor of 1,000 is bounding of the sensitivity studies recommended in the EPRI guidance and therefore is acceptable. Assuming an increase in corrosion likelihood of 1,000, the increase in LERF is 5.22E-08 per year for Unit 1 and 2.82E-08 per year for Unit 2. These changes are considered to be "very small" (i.e., below 1E-07 per year) per the acceptance guidelines in RG 1.174.

   As indicated in Tables 5-29 and 5-30 of Section 5.3.1 of Attachment 3 to the LAR, revised in response to NRC RAIs (Reference 2), the reported increase in LERF for internal and external events combined is 7.89E-07 per year for Unit 1 and 9.88E-07 per year for Unit 2. The risk contribution from external events includes the effects of internal fires and seismic events based on IPEEE, and high winds, as discussed in Section 3.5.2.1 of this SE. These LERF increases are considered to be "small" (i.e., between 1E-06 per year and 1E-07 per 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 per year. Per revised Section 5.3.1 of Attachment 3 to the LAR, the total LERF, including internal and external events, is estimated to 8.26E-06 per year for Unit 1 and 1.03E-05 per year for Unit 2 when using IPEEE fire and seismic results. The total LERF for Unit 1, given the increase in ILRT interval, is below 1E-05 per year. The licensee clarifies that the Unit 2 seismic LERF is between 3.69E-07 per year and 1.66E-06 per year and that the highest seismic LERF value was conservatively used to calculate the total LERF. Use of the highest seismic risk value in the range indicates that a reasonable estimate would be somewhat lower, most likely reducing the LERF estimate for Unit 2 below 1E-05 per year. The licensee also stated that, "it is reasonable to conclude that the total Unit 2 LERF is less than 1.0E-5." The NRC staff finds that the LERF is expected to be less than 1E-5 per year. Therefore, the NRC staff agrees that the estimated risk increase associated with permanently extending the ILRT interval is small, consistent with the guidance in RG 1.174.

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 of 0.20 person-rem per year for Unit 1 and 0.11 person-rem per year for Unit 2. These values, reported in Section 3.3.3 of Attachment 1 to the LAR, do not include the steel liner corrosion effects. Assuming an increase in corrosion likelihood of 10,000, the increase in population dose is 0.36 person-rem per year for Unit 1 and 0.20 person-rem per year for Unit 2, as shown in Tables 5-35 and 5-36 of revised Attachment 3, provided by the licensee in response to NRC RAIs (Reference 2). The reported values are 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 integrated plant risk 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 reported by the licensee in Section 3.3.3 of Attachment 1 to the LAR as 0.558 percent for Unit 1 and 0.490 percent for Unit 2. However, Section 5.2.5 of Attachment 3 to the LAR reports increases in CCFP as 0.876 percent for Unit 1 and 0.871 percent for Unit 2, which are based on conservative Class 1 frequency values that include scenarios with successful containment spray or late releases. Assuming an increase in steel liner corrosion likelihood of 10,000, the increase in CCFP is 0.956 percent for Unit 1 and 0.950 percent for Unit 2, as shown in Tables 5-33 and 5-34 of revised Attachment 3, provided by the licensee in response to NRC RAIs (Reference 2). The reported values are below the acceptance guidelines in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

Based on 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 that the increase in the total integrated plant risk and the magnitude of the change in the 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 the three quantitative risk metrics 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.

3.5.4 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 3.3.1 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 (i.e., accident case 3b), and this value has been used in the CCNPP plant-specific risk assessment. Accordingly, the third condition is met.

3.5.5 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 3.3.1 of Attachment 1 to the LAR, the licensee stated that for CCNPP containment over-pressure is not relied upon for ECCS performance. Accordingly, the fourth condition is met.

3.5.6 PRA Technical Evaluation Conclusions

Based on the above, the NRC staff concludes that the LAR for a permanent extension of the Type A containment ILRT frequency from three in 10 years to once in 15 years, for CCNPP, is acceptable.

3.6 Evaluation of Proposed Changes to TS 5.5.16

The CCNPP TS 5.5.16, "Containment Leakage Rate Testing Program," currently states in part:

> A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:
>
> a. Nuclear Energy Institute (NEI) 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. The first Unit 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016.
>
> b. Unit 1 is excepted from post modification integrated leakage rate testing requirements associated with steam generator replacement.

c. Unit 2 is excepted from post modification integrated leakage rate testing requirements associated with steam generator replacement.

In its letters dated September 18, 2014 (Reference 1) and February 17, 2015 (Reference 2), the licensee proposed to remove exceptions (a), (b), and (c) and 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 5.5.16 to state, in part:

> A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

Based on the NRC staff's evaluation in this SE, the licensee has demonstrated that the proposed test intervals are acceptable, and meet the applicable conditions and limitations in NEI 94-01, Revision 2-A (Reference 6), as well as the prescribed guidelines in Revision 3-A (Reference 5). Therefore, the NRC staff concludes that the above change is acceptable.

3.7 Technical Evaluation Conclusions

Based on the considerations discussed above, the NRC staff concludes that the proposed TS changes regarding the frequency of the 10 CFR Part 50, Appendix J, Type A and Type C tests are acceptable.

The next Type A test for CCNPP Unit 1 may therefore be conducted no later than May 3, 2021, instead of the current due date of May 3, 2016; and the next CCNPP Unit 2 Type A test may be conducted no later than March 17, 2028, instead of the current due date of March 17, 2023.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the 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 published in the *Federal Register* on November 25, 2014 (79 FR 70214). 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 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. Gellrich, George H., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-53 and DPR-60, License Amendment Request: Revise Technical Specification Section 5.5.16 for Permanent Extension of Type A and C Leak Rate Test Frequencies," September 18, 2014, ADAMS Accession No. ML14265A219.

2. Gellrich, George H., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Renewed Facility Operating License Nos. DPR-53 and DPR-60, "Request for Additonal Information Regarding Permanent Extension of Type A and C Leak Rate Test Frequencies License Amendment Request," February 17, 2015, ADAMS Accession No. ML15051A409.

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Principal Contributor(s): A. Sallman, NRR
M. Biro, NRR
F. Farzam, NRR

Date: July 16, 2015

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

Sincerely,

/RA/

Alexander N. Chereskin, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 310 to DPR-53
2. Amendment No. 288 to DPR-69
3. Safety Evaluation

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M. Brio, NRR
F.Farzam, NRR
A. Sallman, NRR

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DATE	6/08/2015	6/08/2015	5/13/2015	4/27/2015	5/26/2015
OFFICE	DSS/STSB/BC	OGC	DORL/LPLI-1/BC(A)	DORL/LPLI-1/PM	
NAME	RElliott*	JWachutka	MDudek	AChereskin	
DATE	6/10/2015	6/17/2015	7/16/2015	7/16/2015	

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