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

March 22, 2011

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2 - AMENDMENT RE: ONE-TIME 5-YEAR EXTENSION TO THE CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL (TAC NO. ME4804)

Dear Mr. Gellrich:

The Commission has issued the enclosed Amendment No. 274 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated October 4, 2010, as supplemented by letter dated December 9, 2010.

The amendment will revise TS 5.5.16, "Containment Leakage Rate Testing Program," to allow a one-time 5-year extension of the containment integrated leak rate test (CILRT) interval from 10 to 15 years. This will require the licensee to perform its next CILRT no later than May 1, 2016.

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

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-318

Enclosures:
1. Amendment No. 274 to DPR-69
2. Safety Evaluation

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

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 274
Renewed License No. DPR-69

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

   A. The application for amendment by Calvert Cliffs Nuclear Power Plant, Inc., (the licensee) dated October 4, 2010, as supplemented by letter dated December 9, 2010, 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:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 274, are hereby incorporated into the renewed 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 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado, 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: March 22, 2011

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 274 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following page of the 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 page of the Appendix A Technical Specifications 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
5.5-17	5.5-17

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

The licensee 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. 274 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 radiological monitoring program specified in the Appendix B Technical Specifications, the licensee will provide to the staff a detailed analysis of the problem and a program of remedial action to be taken to eliminate or significantly reduce the detrimental effects or damage.

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 Regulatory Guide 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.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-69

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-318

1.0 INTRODUCTION

By application dated October 4, 2010 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML102800480), as supplemented by letter dated December 9, 2010 (ADAMS Accession No. ML103470280), Calvert Cliffs Nuclear Power Plant, LLC, the licensee, requested changes to the Technical Specifications (TSs) for Calvert Cliffs Nuclear Power Plant (CCNPP). The proposed change would revise TS 5.5.16, "Containment Leakage Rate Testing Program," to allow a one-time extension to the 10-year frequency for the next containment integrated leak rate test (CILRT), or Type A test, at CCNPP Unit No. 2. The proposed change would permit the existing CILRT frequency to be extended, on a one-time basis, from 10 years to 15 years.

The containment leak rate testing program requires the licensee to perform a CILRT and local leak rate tests (LLRTs) called Type B and Type C tests. The Type A test measures the overall leakage rate of the primary reactor containment. Type B tests are primarily intended to detect leakage paths and measure leakage rates for the primary reactor containment penetrations. Type C tests are intended to measure containment isolation valve leak rates.

The current 10-year CILRT interval for Unit No. 2 ends on May 1, 2011. Approval of the licensee's request will allow CCNPP to defer the next CILRT test to no later than May 1, 2016.

Specifically, the proposed amendment would revise TS 5.5.16.a. to add the following exception:

> 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.

The letter dated December 9, 2010, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. CCNPP Unit No. 2 TS 5.5.16 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program." RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 1, December 2005, "Risk Impact Assessment to Extended Integrated Leak Rate Testing Intervals."

Section 9.2.3.1 of NEI 94-01, Revision 2-A (October 2008), "Industry Guideline for Implementing Performance-Based Option of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J," states that plant-specific confirmatory analyses of the risk associated with integrated leakage rate test (ILRT) interval extensions are required when extending the interval beyond 10 years. Section 9.2.3.4 of NEI 94-01 states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A (October 2008), "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

RG 1.163, Section C, "Regulatory Position" states; "licensees intending to comply with Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 rather than using test intervals specified in American National Standards Institute/American Nuclear Society Standard (ANSI/ANS)-56.8-1994." The industry guidelines in NEI 94-01 state that Type A testing shall be performed at a frequency of at least once every 10 years.

3.0 TECHNICAL EVALUATION

The licensee's proposal would extend the CILRT interval, on a one-time basis, from 10 to 15 years. The licensee justifies the proposed change based upon historical plant-specific Type A test results, containment in-service inspection (CISI) results, and a risk-informed analysis. The risk-informed analysis was performed in accordance with the staff guidance found in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

In support of the defense-in-depth philosophy discussed in RG 1.174, the licensee provided a non-risk based assessment of the multiple inspections and test programs in place at CCNPP that ensure the containment structure remains capable of meeting its design functions. This assessment addresses the current condition of the structural and leak-tight integrity of the CCNPP containment structure and the ability of the licensee's LLRT program and in-service inspection (ISI) program to detect and manage aging degradation of the containment so that the

structural and leak-tight integrity of the containment will be maintained, if the CILRT test interval is extended as proposed. There are no changes to any Code or regulatory requirements.

3.1 Containment Building Description

CCNPP Unit No. 2 is a pressurized-water reactor (PWR) with a steel-lined reinforced concrete primary containment structure, which consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a reinforced concrete foundation slab. The interior surface of the structure is lined with a 1/4" thick welded steel plate to assure a high degree of leak tightness. The containment structure has personnel and equipment access openings as well as numerous mechanical and electrical systems that penetrate the containment structure through welded steel penetrations. The penetrations and access openings are designed, fabricated, inspected, and installed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Class B. The containment structure, in conjunction with Engineering Safeguards Features, is designed to withstand the internal pressure and coincident temperature resulting from the energy released in the event of a postulated loss-of-coolant-accident (LOCA) associated with rated full power operation. The design conditions for the containment structure include an internal pressure of 50 psig and a coincident concrete surface temperature of 276 °F. The maximum allowable containment leakage rate is 0.16 percent of containment air weight per day at the maximum calculated containment internal pressure of 49.4 psig.

The containment pressure boundary consists of the steel liner, containment access penetrations, and the penetrations for process piping and electrical wiring. The integrity of the penetrations and containment isolation valves is verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J, and the overall integrity of the containment structure is verified through a Type A test. These tests are performed to verify the leak-tight integrity of the containment structure at the design-basis accident (DBA) pressure. The leak rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Type A, Type B and Type C tests) and the CISI requirements, required by 10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment structure during its service life.

3.1.1 Current CCNPP CILRT Requirements

TS 5.5.16 states:

> The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 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.

The maximum allowed containment leakage rate, L_a, specified in TS 5.5.16, ensures that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. As an added conservatism to account for possible degradation of the containment leakage barriers between leakage tests, the leakage acceptance criteria under TS 5.5.16.a is limited as follows:

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

3.2 Historical Testing Results

3.2.1 CCNPP Unit No. 2 Type A Tests

The two most recent CILRTs were performed on January 16, 1991, and May 2, 2001. The current test interval would require that the next CILRT be performed no later than May 1, 2011.

The results of the last two Type A CILRTs are as follows:

Test Date	January 16, 1991	May 2, 2001	Maximum allowable
Total As-Found Leakage	0.061 %/day	0.0738 %/day	≤ 16 %/day
Total As-Left Leakage	0.001 %/day	0.0014 %/day	≤ 16 %/day

Both tests were successful because the measured containment leakage was within the acceptance limits. In addition, an ample margin indicated a leak-tight containment structure. With the requested 5-year extension of the CILRT interval, the next test would be performed no later than May 1, 2016.

3.2.2 CCNPP Unit 2 Type B and Type C Testing Program

The Type B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B, and RG 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per TS 5.5.16, the allowable maximum pathway total for Type B and C leakage is 0.6 La. (La equals approximately 276,800 sccm).

Tables 1 and 2 provide the local leak rate test (LLRT) data trend summaries for Calvert Cliffs Unit No. 2, since the performance of the 2001 CILRT. This summary shows that there has been no as-found failures that resulted in exceeding the TS 5.5.16 limit of 0.6 La (166,080 sccm) and, therefore, demonstrates a history of successful tests.

Table 1.
Unit 2 Type B and C LLRT Combined As-Found (AF)/As-Left (AL) Trend Summary

RFO	2003	2005	2007	2009
AF MAX PATH (sccm)	12051.84	15759.7	14943.1	26859.7
Fraction of La	0.035	0.046	0.043	0.078
AF MIN PATH (sccm)	10535.95	14380.4	10689.8	14570.3
Fraction of La	0.030	0.042	0.031	0.042
AL MAX PATH (sccm)	12347.1	3848.9	13936.6	11969.8
Fraction of La	0.036	0.011	0.040	0.035
AL MIN PATH (sccm)	11091.6	2784.9	9070.2	7028.9
Fraction of La	0.032	0.008	0.026	0.020

The U.S. Nuclear Regulatory Commission (NRC) staff requested the licensee to clarify the following concerns relative to the above Table 1:

- The licensee was requested to explain why the as-left (AL) leakage was greater than the as-found (AF) leakage in 2003. The licensee attributed the results to penetrations 2B and 21SG.

  Regarding penetration 2B, a check valve was AF tested within the administrative limit of 296 standard cubic centimeters per minute (SCCM). However, the valve was replaced during the 2003 outage. The AL leakage of the new valve was 1488 SCCM. This was significantly higher than the administrative limit but lower than the maximum limit of 10000 SCCM. The licensee subsequently removed penetrations 2 and 8 from the Appendix J program scope and the Updated Final Safety Analysis Report, based on the determination that they met the criteria of a "water filled penetration" and therefore not subject to leakage rate testing per 10 CFR Part 50, Appendix J. Therefore, there are no further records of leakage for these penetrations since 2003.

  Regarding penetration 21SG (21 SG South Manway), the AF leakage rate was significantly less than the administrative limit. The manway covers were removed during the 2003 refueling outage (RFO) to facilitate replacement of the steam generators. The AL test showed a significantly greater leakage than the AF value, but lower than the administrative limit. The manways were removed and reinstalled during subsequent outages and the results continue to be below the administrative limits.

- The licensee was requested to explain why the AF value in the most recent test in 2009 was significantly greater than previous tests. The licensee attributed the results to penetrations 44 and 61.

  Regarding penetration 44 (fire protection piping), AF leakage rate from a check valve was

measured at 7480 SCCM, well above the administrative limit of 887 SCCM, but below the maximum leakage limit of 20000 SCCM. The valve was disassembled for inspection and flushing. No significant adverse conditions were found. The probable cause for increased leakage was considered to be due to debris buildup. The AL value following maintenance was 744 SCCM, a significant improvement from the AF value.

Regarding penetration 61 (spent fuel pool cooling), both containment isolation valves were AF tested, resulting in an AF max pathway leakage value of 5000 SCCM, significantly higher than the administrative limit of 1182 SCCM but below the maximum leakage limit of 20000 SCCM. A maintenance work order was performed to re-torque and cycle the applicable valves. The AL value of 86 SCCM showed significant improvement.

Table 2 identifies the number of Type B and C LLRTs which were found to exceed their administrative limits (assigned limit that is less than the TS limit), which results in decreasing the intervals between subsequent LLRTs for that component.

Table 2.
Unit 2 As-Found LLRTs Exceeding Admin Limit Summary

RFO	2003	2005	2007	2009
Number AF LLRTs Exceeding Admin Limit	2 Type C 0 Type B	3 Type C 0 Type B	5 Type C 0 Type B	8 Type C 0 Type B

The NRC staff requested the licensee to clarify the following concerns relative to the above Table 2:

- The NRC staff noted an increasing trend in the number of LLRTs exceeding the administrative limits and requested the licensee to provide details as to where they occurred and what was done to correct the problem.

  The licensee stated that when the results from 1997 onwards are taken into consideration (instead of 2003 onwards as reflected in Table 2 above), there is no appreciable negative trend. The licensee provided additional information to show that the combined leakage rate for those pathways with components whose leak rates are in excess of the administrative limits reflects a relatively flat trend from 2001 onwards. The licensee further stated that the slight increase in AF LLRT rate between 2007 and 2009 RFOs is explained by the penetrations 44 and 61 discussed above. The licensee also claimed that the administrative leakage limits at CCNPP are generally more conservative than used by the industry by a factor of up to 10 times.

- The NRC staff noted in the licensee's application that leakage through penetration 48B (hydrogen purge supply) had significantly increased in the most recent tests in 2007 and 2009, and requested the licensee to provide the reasons for this increase and the actions taken or being taken to correct this trend.

  The licensee stated that during the 2007 RFO, penetration 48B AF leakage rate was 2600 SCCM, which was above the administrative limit of 591 SCCM, but below the maximum

leakage limit 10000 SCCM. A corrective action was initiated to document and trend the issue, but a decision was made not to perform maintenance as the leakage did not challenge the maximum limit and there were no outstanding hardware related corrective actions on the subject valve. The subsequent LLRT of this valve in the 2009 RFO indicated that AF leak rate remained approximately the same as in the 2007 RFO. A decision was made at that time to perform maintenance on the valve during the 2011 RFO and a maintenance work order to overhaul the valve is currently on the 2011 RFO schedule.

Based on the NRC staff review of the information provided by the licensee, with the exception of 1985, all CILRT results were less than 60% of the performance criterion value (0.75% La). There are no apparent adverse trends that would suggest containment leakage potential would exceed L_a during the requested 5-year interval extension. The Type B and C AF minimum pathway totals were less than 16% of their performance criterion (0.6La) and were also without any apparent adverse trend to suggest containment leakage potential would exceed L_a during the requested 5-year interval extension. The Type B and Type C testing schedules are expected to be minimally impacted by the requested CILRT extension and will continue to be performed and results totaled each RFO. Penetration leakage is expected to be the major contributor of any potential containment leakage and the Type B and Type C tests will continue to allow monitoring of potential penetration leakage at the existing allowed intervals for these tests.

3.3 Containment Inservice Inspection Program

The Calvert Cliffs' CISI program periodically performs destructive and nondestructive examinations of ASME Class MC and CC components in order to identify the presence of any service-related degradation. The CISI program is established in accordance with 10 CFR 50.55a. This program has been developed to comply with the ASME Code, Section XI, 2004 Edition, except where specific written alternatives from Code requirements have been requested by Calvert Cliffs and granted by the NRC.

The program defines the Class MC and CC components and the Code-required examinations for each ASME Code, Section XI examination category, and the augmented inspection scope, as applicable. The components subject to the requirements of this CISI program are those which make up the containment structure, its leak tight barrier (including integral attachments) and those which contribute to its structural integrity, specifically, Class MC pressure-retaining components, and their integral attachments and Class CC post tensioned concrete containments.

3.3.1 IWE (Class MC) Inspection Interval and Periods

The second 10-year containment ISI interval for both units for the performance of CISI (IWE) complies with IWE-2412 Inspection Program B and began on September 9, 2009, and will end on September 9, 2018. This interval is shortened as a result of extending the first 10-year

containment ISI interval by 1 year. The interval is then further divided into three periods which are, as follows:

1st Period: September 9, 2009 through September 9, 2011 (2 years)
2nd Period: September 9, 2011 through September 9, 2015 (4 years)
3rd Period: September 9, 2015 through September 9, 2018 (3 years)

3.3.2 IWL (Class CC) Inspection Periods (Concrete)

The second 10-year containment interval for the performance of CISI (IWL) for both units complies with IWL-2400 and is effective for IWL inspections conducted between September 9, 2009 and September 9, 2018.±

Concrete examinations shall be conducted every 5 years (± 1 year), as described in IWL-2410 (a) and (c). For the purpose of the CISI program, an IWL inspection period is 5 years, with two periods per inspection interval.

Concrete surface areas affected by a repair/replacement activity shall be examined at 1 year (± 3 months) following completion of repair/replacement activity. If plant operating conditions are such that examination of portions of the concrete cannot be completed within this time interval, examination of those portions may be deferred until the next regularly scheduled plant outage.

3.3.3 IWL (Class CC) Inspection Periods (Tendons)

For multiple-unit plant sites, such as Calvert Cliffs, the tendon examination frequency may be extended to 10 years per unit, provided the containment structures utilize the same pre-stressing system, are essentially identical in design, had their original structural integrity test performed within 2 years of one another, and experience similar environmental exposure. The examinations required by IWL-2500 for unbonded post-tensioning systems can then alternate between the two units every 5 years, as allowed by IWL-2421 (sites with multiple units).

Going forward for Calvert Cliffs Unit No. 2, the following two ASME required tests are to be performed once every 10 years.

- Tendon force and elongation measurements (tendon lift-off test)
- Tendon wire and strand sample examination and testing (wire removal tensile test)

These tests are scheduled to be performed no later than 2013.

The following three ASME required tests are to be performed once every 5 years.

- Examination of tendon anchorage areas (visual examination)
- Sampling and analysis of corrosion protection medium (grease analysis)
- Free water collection and analysis (free water analysis)

These tests are also scheduled to be performed no later than 2013. The table below shows the timeline for the IWE/IWL inspections during the second 10-year containment ISI interval.

CCNPP Unit 2 IWE/IWL Examination Periods

Period	Date	Tolerance
35 year	9/9/2012	+/- 1 Year
40 Year	9/9/2017	+/- 1 Year

In summary, for the 15-year extended CILRT interval, the containment structure will have at least three visual examinations (one already performed in 2009, and one planned in 2011 and 2015) prior to performance of the pre-CILRT visual examination in 2016.

The NRC staff finds the three additional visual examinations prior to the pre-CILRT visual examination for a 15-year interval, which is consistent with Regulatory Position C.3 of RG 1.163, acceptable.

On the basis of its review of the information provided in the licensee's TS change request, the NRC staff finds that: (1) the results of previous CILRTs demonstrate that the leak-tight integrity of the containment structure has been adequately managed; (2) the structural integrity of the containment vessel is verified through periodic ISI conducted as required by Subsections IWE and IWL of the ASME Code, Section XI; (3) the integrity of the penetrations and containment isolation valves are periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J, and CCNPP TS; (4) the licensee is employing a CISI program that requires evaluation of any potential degradation of inaccessible areas of the containments, and (5) the containment liner protective coating is visually inspected every refueling outage and repair of any identified damage is adequately managed. Based on these findings, the staff concludes that the licensee has an adequate ISI program and procedures in place to examine, monitor, and correct potential age-related and environmental degradations of the pressure retaining components of the CCNPP Unit 2 containment structure. Therefore, the licensee's proposed one-time extension of the CILRT interval from 10 to 15 years is acceptable.

3.4 Risk Analysis

The licensee performed a risk impact assessment of extending the Type A test interval from 10 to 15 years. The risk assessment was provided in the October 4, 2010, application for license amendment. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in EPRI Topical Report (TR)-1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," October 2008, and NRC RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 (ADAMS Accession No. ML003740133).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement

this basis, the industry undertook a similar study; the results of that study are documented in EPRI TR-104285. The EPRI-TR-104285 study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for CCNPP Unit 2 early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests, based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of pre-existing containment leakage. The risk contribution of pre-existing leakage for the PWR and boiling-water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 10 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person roentgen equivalent man (rem) per year in increased public dose.

The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance and the EPRI-TR-1018243 report to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 states that a PRA used in risk-informed regulation should be performed in a manner that is consistent with accepted practices. In NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007, ADAMS Accession No. ML070650428), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated January 2007 (ADAMS Accession No. ML070240001), to determine whether the technical adequacy of the PRA used to support a submittal is consistent with accepted practices. Revision 2 of RG 1.200 will be used for all risk-informed applications received after March 2010. In the Final Safety Evaluation for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105), the NRC staff states that Capability Category I of the ASME PRA Standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their contribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

3.4.1 Technical Adequacy of the PRA

The licensee's October 4, 2010 license amendment request addresses the technical adequacy of the PRA that forms the basis for the subject risk assessment. The CCNPP Unit 2 PRA internal events model meets ASME PRA Standard RA-Sb-2005 Capability Category II and RG 1.200, Revision 2. An industry peer review team reviewed the CCNPP Unit 2 PRA model in June 2010. As part of the ILRT extension application, the licensee provided a list of findings that were relevant to the ILRT analysis. A summary of the findings from the peer review, and an assessment of the impact of these findings on the risk assessment for the ILRT extension, are provided in the licensee's submittal dated October 4, 2010. The NRC staff reviewed this

information and has no objection to the conclusions in the licensee's assessment. Given that the licensee has evaluated its PRA against RG 1.200 and the ASME PRA Standard, evaluated all of the findings developed during the reviews of its PRA for applicability to the ILRT extension, and determined that any unresolved issues would not impact the conclusions of the ILRT risk assessment, the NRC staff concludes that the current CCNPP Unit 2 PRA model is of sufficient technical quality to support the evaluation of changes to ILRT frequencies.

3.4.2 Estimated Risk Increase

RG 1.174 provides risk-acceptance guidelines for assessing the increases in CDF and LERF for risk-informed license amendment requests. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed amendment based on the cumulative change from the original frequency of three tests in a 10-year interval, the current test interval of 10 years, and the proposed testing interval of 15 years. RG 1.174 also discusses defense-in-depth. The licensee estimated the change in the conditional containment failure probability for the proposed amendment and judged it to be insignificant and reflecting sufficient defense-in-depth.

The licensee comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A test frequency:

1. Given the change from the 3 in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be 1.57E-1 person-rem per year a 0.438% increase over the total baseline risk. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the current 3 in 10 years to one in 15 years is estimated to be about 2.59×10^{-8} per year, based on the plant-specific internal events PRA, and about 7.76×10^{-8} per year, when external events are included.

   Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 3 in 10-year to one in 15-year is therefore considered non-risk significant and the results support this determination. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also discusses the need to show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The

licensee estimates the change in the conditional containment failure probability to be an increase of approximately 1 percentage point for the cumulative change of going from a test frequency of 3 in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on the above findings, the NRC staff concludes that the increase in projected risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174, and is, therefore, acceptable.

Based on the above, the NRC staff concludes that the proposed license amendment request for a one-time, 5-year extension of the Type A containment integrated leak rate test interval for the CCNPP Unit 2, is acceptable. In accordance with revised TS 5.5.16, the next Type A tests for CCNPP Unit 2, shall be performed no later than May 1, 2016.

Based on the foregoing evaluation, the staff finds that there are no significant increases in risk or reductions in safety resulting from the requested test extension. Furthermore, the CCNPP Unit No. 2 containment has a reasonably good recent leakage rate history. Therefore, the staff concludes that the requested TS change, increasing the Type A test interval one-time to 15 years, is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (76 FR 1646). Accordingly, the amendment meets 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 amendment.

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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Calvert Cliffs Nuclear Power Plant, LLC letter dated October 4, 2010 "License Amendment Request: One-Time Extension of the Containment Integrated Leak Rate Test Interval" (ML102800480).

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Date: March 22, 2011

March 22, 2011

Mr. George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2 - AMENDMENT RE: ONE-TIME 5-YEAR EXTENSION TO THE CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL (TAC NO. ME4804)

Dear Mr. Gellrich:

The Commission has issued the enclosed Amendment No. 274 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated October 4, 2010, as supplemented by letter dated December 9, 2010.

The amendment will revise TS 5.5.16, "Containment Leakage Rate Testing Program," to allow a one-time 5-year extension of the containment integrated leak rate test (CILRT) interval from 10 to 15 years. This will require the licensee to perform its next CILRT no later than May 1, 2016.

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

Sincerely,
/ra/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-318

Enclosures:
1. Amendment No. 274 to DPR-69
2. Safety Evaluation

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