May 1, 2002

Mr. Charles H. Cruse
Vice President - Nuclear Energy
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. 1 - AMENDMENT RE: ONE-TIME EXTENSION OF APPENDIX J, TYPE A, INTEGRATED LEAK RATE TEST INTERVAL AND EXCEPTION FROM PERFORMING A POST-MODIFICATION TYPE A TEST (TAC NO. MB3929)

Dear Mr. Cruse:

The Commission has issued the enclosed Amendment No. 252 to Renewed Facility Operating License No. DPR-53 for the Calvert Cliffs Nuclear Power Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated January 31, 2002, as supplemented by letter dated March 27, 2002.

The amendment allows a one-time 5-year extension, for a total of 15 years, for the performance of the next Unit 1 integrated leak rate test (ILRT). The amendment also exempts Unit 1 from the requirement to perform a post-modification containment ILRT associated with the steam generator replacement.

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/

Donna Skay, Project Manager, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-317

Enclosures: 1. Amendment No. 252 to DPR-53
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

Accession Number: ML021080753
*Input provided by safety evaluation dated April 3, 2002, incorporated with no significant changes.
** Input provided by safety evaluation dated April 16, 2002,incorporated with no significant changes.
*** See previous concurrence

OFFICE	PDI-1/PM	PDI-1/LA	PDI-1/SC	BC:EMEB	SC:SPLB/ SPSB	OGC
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DATE	4/29/02	4/29/02	4/30/02	04/03/02	04/16/02	04/24/02

OFFICIAL RECORD COPY

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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DATED: May 1, 2002

AMENDMENT NO. 252 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53
CALVERT CLIFFS UNIT 1

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CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 252
Renewed License No. DPR-53

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 January 31, 2002, as supplement by letter dated March 27, 2002, 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. 252, are hereby incorporated in the 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 1, 2002

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 252 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

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
5.0-30	5.0-30
5.0-31	5.0-31

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 252 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-53

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 INTRODUCTION

By letter dated January 31, 2002, as supplemented March 27, 2002, the Calvert Cliffs Nuclear Power Plant, Inc. (CCNPPI or the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit No. 1 (CCNPP), Technical Specifications (TSs). The requested change would allow a one-time 5-year extension, for a total of 15 years, for the performance of the next Unit 1 integrated leak rate test (ILRT). The amendment would also allow an exception from the requirement to perform a post-modification containment ILRT associated with the Unit 1 steam generator replacement. The March 27, 2002, supplemental letter provided clarifying information that did not change the scope of the original notice or the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

2.1 Type A Test Interval Extension

Calvert Cliffs Unit 1 TS 5.5.16 requires 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 Part 50, Appendix J, Option B. 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. A Type A test is an overall ILRT of the containment structure.

The CCNPP TS 5.5.1.6 further requires that the leakage rate testing program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, including errata, as modified by exceptions set forth in the TSs. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances.

The licensee has performed five ILRTs on Unit 1 since the pre-operational Type A test was completed on December 1, 1973. These five ILRTs were completed on March 6, 1978, June 22, 1982, May 20, 1985, May 27, 1988, and July 5, 1992. Based on the last two successful Type A tests at CCNPP and the requirements of 10 CFR Part 50, Appendix J, Option B, the current interval requirement is 10 years.

The licensee is requesting an addition to TS 5.5.16, “Containment Leakage Rate Testing Program,” which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, that 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.

2.2 Exception From Performing a Post-Modification Type A Test

NEI 94-01, Revision 0, Section 9.2.4, “Containment Repairs and Modifications,” states:

> Repairs and modifications that affect containment integrity require leakage rate testing (Type A testing or local leakage rate testing) prior to returning the containment to operation.

The licensee intends to replace the existing Unit 1 steam generators during the spring 2002 refueling outage. The steam generator replacement affects only the closed piping inside containment. The containment structure and the containment liner are not affected. The new steam generator assemblies and the old steam generator assemblies will transit through the containment equipment hatch. However, the steam generator shell and the inside-containment portions of the main steam, feedwater, steam generator blowdown, and auxiliary feedwater lines are part of the primary reactor containment boundary that will be impacted by the replacement activities.

The licensee is requesting an addition to TS 5.5.16, “Containment Leakage Rate Testing Program,” which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, that Unit 1 be excepted from post-modification ILRT requirements associated with steam generator replacement.

3.0 EVALUATION

3.1 Inservice Inspection Program

The licensee stated that the inservice inspection (ISI) of the CCNPP containment building is conducted in accordance with the requirements of the 1992 Edition with 1992 Addenda of Subsections IWE and IWL of ASME Section XI, including the NRC-approved requests for relief from certain Code requirements. The licensee also performs, under its Safety-Related Protective Coatings Program at CCNPP, a walkdown of the containment interior at the beginning of each refueling outage to determine areas requiring repair. The results of these examinations indicated that there are some minor areas of coating degradation occurring, but nothing significant that would adversely impact either the containment structural integrity or its leak tightness. All identified areas of minor coating degradation were evaluated and found to be limited in scope, with no significant liner material loss, and no potential for precursors to significant containment liner failures.

The licensee stated that several areas were identified as candidate areas for augmented examination (required by IWE Table-2500-1, Examination Category E-C), in accordance with IWE-1241. These included areas beneath the liner to floor slab moisture barriers, potential ponding areas at structural steel attachments, and several areas with photographic evidence of dark areas. The evaluation of these identified areas reached the following conclusions:

1. No ponding areas were evident either as being presently wet or by the presence of watermarks.

2. The dark areas were identified in both cases to be insulation at a penetration.

3. The area beneath the moisture barrier was found to suffer from scaling, rust, and pitting. Areas visually representative of the worst of these were selected for detailed examination and documented using a combination of ultrasonic thickness measurement, pit depth measurement, and detailed visual examination. These areas are now designated as augmented examination areas in accordance with American Society of Mechanical Engineers (ASME) Subsection IWE, and are subject to repeated examination once per ISI period as required by Subsection IWE.

The Nuclear Regulatory Commission (NRC) staff finds that the licensee's ISI program, including areas of augmented inspections, will provide adequate assurance that the containment structural integrity will be maintained during the extended ILRT period.

The licensee stated that the containment leak-tight integrity is also verified by local leak tests (Type B test) of containment penetration bellows, airlocks, seals, and gaskets. The alternate examinations of Appendix J Type B testing for seals and gaskets, as approved in relief request E1, will be performed at least once during each containment inspection interval. Thus, the one-time extension requested for Type A testing does not affect the frequency of these alternate examinations in that they will be performed once in the third 10-year inspection interval. For the bolted connections, the licensee stated that examinations required by ASME Table IWE-2500-1, Categories E8.10 and E8.20 are performed during preventive maintenance activities of certain components. These maintenance activities are scheduled to support replacement of the seals and gaskets used in the component connections. Additionally, some of these connections are routinely used during outages, and the examination and testing of these connections are performed to re-establish containment integrity at the end of the outage. Any parts (except for seals and gaskets, which are exempt) that are replaced are subject to compliance with CCNPP Repair and Replacement Program and receive the appropriate inspections at that time. The staff finds that the licensee's ISI program for seals, gaskets, and bolted connections provides reasonable assurance that the integrity of the containment pressure boundary will be maintained.

3.2 Risk Assessment

The licensee performed an assessment of the risk associated with extending the Type A test interval to 15 years. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in the Electric Power Research Institute

(EPRI) TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” and RG 1.174, “An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

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 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 support rulemaking 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 the NRC’s rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI 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 EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time that a leak 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 leakage. This increase in the probability of leakage results in an increase in the contribution of pre-existing leaks to the predicted person-rem/year frequency. In terms of increased public dose (person-rem\year), the increase ranged from 0.02 to 0.14 percent. The EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an “imperceptible” increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Based on the licensee’s submittal, the increase in risk in terms of person-rem/year in going from the original 3 in 10-year test interval to the current 1 in 10-year test interval was 0.16 percent. The NRC staff finds this value comparable to the upper range of that estimated in NUREG-1493.

Since the Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant’s licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. 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 change and the cumulative change from the original 3 in 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics. The methodology is consistent with previously approved similar requests.

The NRC staff has drawn the following conclusions based on its analysis of the change in risk associated with extending the Type A test frequency:

1. An increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.08 percent. The increase in the total integrated plant risk, given the change from a 3 in 10-year test interval to a 15-year test interval, was 0.24 percent. The increase in total integrated plant risk, given the change from a 3 in 10-year test interval to the current 1 in 10-year test interval, was 0.16 percent. This is comparable to the increase in risk estimated in NUREG-1493 in reducing the frequency of tests from 3 per 10 years to 1 per 10 years. 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 interval from the original 3 in 10 years to 1 in 15 years is estimated to be 2.9×10^{-7}/year using a methodology based on the EPRI study.

   However, there is some likelihood that the undetected flaw in the containment liner estimated as part of the class 3B frequency would be detected as part of the IWE visual examination process of the containment liner. The containment was visually inspected in 2000 and 2002. An additional visual inspection is now planned for 2004. Eighty-five percent of the inner containment liner can be visually inspected. If one assumes the visual inspections are 90 percent effective in detecting large flaws in the visible regions of the containment (5 percent for failure to detect and 5 percent for flaw not detectable (not-through-wall)), then the increase in LERF would go from 2.9×10^{-7}/year to 6.8×10^{-8}/year. Therefore, increasing the Type A interval to 15 years is considered to be a very small change in LERF when using the guidelines of RG 1.174.

   The licensee performed additional risk analysis to consider the impact of hypothetical corrosion in inaccessible areas of the containment liner on the proposed change. The inaccessible areas included the backside of the containment liner. The risk analysis considered the likelihood of an age-adjusted liner flaw that would lead to a breach of containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type ILRT. When possible corrosion of the containment liner is considered, the increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 7.8×10^{-8}/year. This additional risk analysis provides added assurance that increasing the Type A interval to 15 years is a very small change in LERF.

3. RG 1.174 also encourages the use of risk analysis techniques to confirm 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 among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to increase by 0.0013 for the proposed change and 0.0031 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.3 Exception from Performing a Post-Modification Type A Test

As discussed in Section 2.0 above, the steam generator shell and the inside-containment portions of the main steam, feedwater, steam generator blowdown, and auxiliary feedwater lines are part of the primary reactor containment boundary that will be impacted by the replacement activities.

The affected area of the containment boundary is also part of the pressure boundary of an ASME Class 2 component/piping system and, as such, the replacement of the steam generators is subject to the repair and replacement requirements of ASME Section XI. The ASME Section XI surface examination, volumetric examination, and system pressure test requirements are, in some ways, more stringent than the Type A testing requirements. The acceptance criteria for ASME Section XI system pressure testing of welded joints is "zero leakage." In addition, the test pressure for the system pressure test will be approximately 17 times that of a Type A test.

The ASME Section XI testing provides an alternative method which allows testing of only the modified portions of the containment barrier (steam generator shell and associated closed piping) instead of the more comprehensive Type A testing which would be performed on the entire containment barrier.

The use of ASME Section XI testing in place of Type A leakage rate testing is not consistent with the current TS 5.5.16, “Containment Leakage Rate Testing Program.” Additionally, the proposed change would accomplish leakage testing of the modified portions of the containment barrier in Mode 3, in contrast to the current requirement to complete testing prior to entering Mode 4. ASME Section XI requires testing to be performed at approximately normal reactor operating temperature and pressure.

The ASME Section XI pressure test, unlike the Type A test, does not require the leakage rate to be quantified. The acceptance criterion for the proposed test is no visual through-wall leakage; therefore, there is no need to quantify the leakage rate. This acceptance criterion is more conservative than the Appendix J Type A test which allows some leakage. However, the ASME Section XI pressure testing can be done without removing the insulation over the piping. This allows some uncertainty in the leakage rate.

ASME Section XI requires non-destructive examination (NDE) and visual examination of welds and system leakage testing. If any through-wall leakage is detected from the welds, the leakage is required to be repaired before plant service continues. NDE of the welds (ultrasonic or radiographic testing) provides assurance that the joints are free of flaws that could result in significant leakage. This NDE provides confidence to pressurize the secondary side of the steam generators and demonstrate leak-tight integrity with the unit in Mode 3 under no-load conditions.

The proposed change for post-modification testing would allow entry into Modes 3 and 4 before the testing is complete because the testing is conducted at normal operating conditions. The licensee asserts that entering Modes 3 and 4 prior to quantifying the containment leakage rate is acceptable because, in order to have a release through the modified closed piping systems, there would have to be a loss-of-coolant accident concurrent with a through-wall leak, with enough pressure in the containment to overcome main steam system pressure.

The ASME Section XI surface examination, volumetric examination, and system pressure testing requirements are more stringent than the Type A testing requirements of Appendix J. The objective of the Type A test is to assure the leak-tight integrity of the containment area affected by the modification. The ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J and the provisions of NEI 94-01, Section 9.2.4. Therefore, the NRC staff finds the basis for the licensee's request to not perform a post-modification Type A test due to the forthcoming steam generator replacements to be acceptable.

3.4 Conclusion

Based on the NRC staff's review of the information provided in the licensee's submittals, the staff finds that (1) the structural integrity of the containment vessel is verified through the periodic inservice inspections conducted as required by Subsections IWE and IWL of the ASME Code, Section XI, (2) 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, and (3) the potential for large leakage from the areas that cannot be examined by the ISI has been explicitly modeled in performing the risk assessment, and (4) the ASME Section XI inspection and testing requirements fulfill the intent of the post-modification Type A testing requirements of Appendix J.

Based on the foregoing evaluation, the NRC staff finds that the interval until the next Type A test at Calvert Cliffs Unit 1 may be extended to 15 years, and that the proposed changes to TS 5.5.16, including the exception from performing a post-modification Type A test due to the forthcoming steam generator replacements, are acceptable.

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 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 (67 FR 7413). 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Cheng
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Date: May 1, 2002