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Constellation
Nuclear

Calvert Cliffs
Nuclear Power Plant

A Member of the
Constellation Energy Group

November 19, 2001

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318
License Amendment Request: Revision to the Containment Leakage Rate Testing Program Technical Specification to Support Steam Generator Replacement

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) hereby requests an amendment to Renewed Operating License No. DPR-69 to incorporate the changes described below into the Technical Specification for CCNPP Unit 2.

DESCRIPTION

Calvert Cliffs Nuclear Power Plant, Inc. will be replacing the Combustion Engineering Model 67 Steam Generators with steam generators fabricated by Babcock & Wilcox Canada Ltd. during the Unit 1 and Unit 2 refueling and replacement outages in the spring of 2002 and 2003, respectively. In accordance with the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, the ten-year Unit 2 containment integrated leakage rate test was successfully performed during the Spring 2001 refueling outage. The proposed amendment would revise the CCNPP Technical Specification 5.5.16 to exempt Unit 2 from the requirements of Appendix J, Option B for post-modification containment integrated leakage rate testing associated with the 2003 steam generator replacement outage. The Unit 1 ten-year integrated leakage rate test is scheduled to be performed during the Spring 2002 replacement outage; therefore, the requirement for post-modification leakage rate testing following Unit 1's steam generator replacement will be met. The detailed descriptions and the safety analyses for the proposed revision is provided in Attachment (1).

REQUESTED CHANGE

Change the CCNPP Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" as shown on the marked-up page in Attachment (3).

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ASSESSMENT AND REVIEW

We have evaluated the significant hazards considerations associated with this proposed change, as required by 10 CFR 50.92, and have determined that there are none (see Attachment 2 for a complete discussion). We have also determined that operation with the proposed modification will not result in any significant change in the types or significant increases in the amounts of any effluents that may be released offsite, and no significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

SAFETY COMMITTEE REVIEW

The proposed amendment to the CCNPP Technical Specification has been reviewed by our Plant Operations and Safety Review Committee and Offsite Safety Review Committee. They have concluded that implementing this amendment will not result in an undue risk to the health and safety of the public.

SCHEDULE

The proposed amendment is scheduled to be implemented following the Unit 2 refueling and replacement outages in spring 2003. To help us prepare for the replacement outage in a timely fashion, we request that you review and approve our application by May 31, 2002, for delayed implementation.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 19th day of November 2001.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: 11/19/01
Date

CHC/GT/bjd

Attachments: (1) Proposed Revision to CCNPP Technical Specifications to Support Steam Generator Replacement; Description and Safety Evaluation
(2) Determination of Significant Hazards
(3) Marked-Up Technical Specification Page

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
D. M. Skay, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

PROPOSED REVISION TO CCNPP TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

Calvert Cliffs Nuclear Power Plant, Inc.
November 19, 2001

ATTACHMENT (1)

PROPOSED REVISION TO CCNPP TECHNICAL SPECIFICATION TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

I. BACKGROUND

Calvert Cliffs Nuclear Power Plant (CCNPP) is a dual unit site. Each unit is a two-loop 2700 MWt Combustion Engineering (CE) design. The original CE Model 67 Steam Generators have been in service since the mid-seventies when the two CCNPP units were first licensed for commercial operation. Calvert Cliffs' is currently preparing to replace the CCNPP steam generators with steam generators fabricated by Babcock & Wilcox Canada Ltd. during the Unit 1 refueling outage in the spring of 2002 (end of Cycle 15), and the Unit 2 refueling outage in the spring of 2003 (end of Cycle 14).

The replacement steam generator (RSG) consists of a new lower subassembly, new steam drum internals, new feedring, with the existing steam drum being refurbished and reattached to the subassembly within the Containment. The RSGs will occupy the same physical envelope as the original steam generators (OSGs). There are no changes to interfaces with the reactor coolant, main feedwater, or main steam systems, or to major component supports or piping supports. Some of the differences between the OSG and RSG designs include: (1) use of thermally-treated Alloy 690 tube material instead of high temperature mill-annealed Alloy 600 used for the OSG; (2) a small weight increase and a small change in the center of gravity location; (3) addition of an integral flow restrictor in the main steam nozzle; (4) increased heat transfer area; and (5) a minor geometric difference in the location of the top of the feed ring with respect to the pedestal.

The CCNPP containment structure consists of a reinforced concrete cylinder and a shallow domed roof that rests on a reinforced concrete foundation slab. The concrete cylinder and dome have a post-tensioned construction design. The post-tensioned design uses several hundred steel wires ("tendons") that run through conduit within the concrete and are placed under tension. This tension produces an external force on the containment structure that would balance the internal forces produced if a loss-of-coolant accident occurred. Attached to the inside of the containment structure is a 1/4 inch thick steel liner. The function of the liner is to provide a leak tight barrier for the Containment. There are three personnel and equipment access openings in the Containment: a two-door personnel air lock, a two-door personnel escape air lock, and a large diameter single door equipment hatch.

The steam generator replacement activities do not affect the containment structure or the actual containment liner. Access for the replacement steam generators as well as removal of the old steam generators will be through the equipment hatch. However, the outer shell of the steam generators, the inside containment portions of the main steam line, the feedwater lines, the steam generator blowdown lines, and the auxiliary feedwater (AFW) lines are all part of the primary reactor containment boundary that will be impacted by the replacement activities. This application revises the CCNPP Technical Specification 5.5.16 to eliminate the requirement to perform unnecessary post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators.

II. DESCRIPTION OF PROPOSED CHANGE

Calvert Cliffs Nuclear Power Plant Technical Specification 5.5.16 states, "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." Regulatory Guide 1.163 (Reference 1) endorses Nuclear Energy Institute (NEI) 94-01, Revision 0 (Reference 2) for methods acceptable to comply with the requirements of Option B. Prior to returning the Containment to operation, NEI 94-01 requires leakage rate testing (Type A testing or local leakage rate testing), following repairs and modification that affect the containment leakage integrity.

In accordance with the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, the ten-year Unit 2 Containment integrated leakage rate test was successfully performed during the Spring 2001 refueling outage. The proposed amendment would exempt CCNPP Unit 2 from the requirements of Appendix J, Option B for post-modification integrated leakage rate testing associated with the 2003 steam generator replacement outage. The Unit 1 ten-year integrated leakage rate test is scheduled to be performed during the Spring 2002 replacement outage; therefore, the requirement for post-modification leakage rate testing following Unit 1's steam generator replacement will be met.

We are proposing a revision to the CCNPP Technical Specification 5.5.16 as shown on the marked-up page in Attachment (3).

III. SAFETY ANALYSIS

The CCNPP Unit 2 plant design incorporates a closed system for transferring steam from the steam generators inside of the primary containment, to the main turbine-generators in the Turbine Building. The inside Containment portion of this closed system consists of the main steam lines, the feedwater lines, the steam generator blowdown lines, the AFW lines, and the outer shell of the steam generators. This closed system inside Containment forms a part of the primary reactor containment boundary.

The planned replacement of the CCNPP Unit 2 steam generators includes the following activities:

- Cutting and removing the main steam and feedwater lines from the steam generators.
- Cutting and removing the upper assemblies of the steam generators.
- Cutting the reactor coolant piping and removing the steam generator lower assemblies.
- Installing the new steam generator lower assemblies and re-welding the reactor coolant piping.
- Re-installing the steam generator upper assemblies on the new lower assemblies.
- Re-installing and re-welding the main steam and feedwater lines.

The planned replacement of the Unit 2 steam generators affects only this closed piping system inside Containment. The steam generator replacement activities do not affect the containment structure or the actual containment liner.

As mentioned above, 10 CFR Part 50, Appendix J, Option B requires integrated leakage rate testing (Type A) or local leakage rate testing (Type B or Type C) prior to returning the Containment to operation following repairs and modification that affect the containment leakage integrity. The Type C testing requirements apply to leakage testing of containment isolation valves. The planned replacement does not affect any containment isolation valves and; therefore, the Type C testing requirements are not applicable. The Type B testing requirements apply to leakage testing of gasketed or sealed containment penetrations (e.g., electrical penetrations), air lock door seals, and other doors with resilient seals or gaskets. Although the secondary side of the steam generators have access manways with gaskets, the Type B testing requirements do not address the other areas of the containment boundary affected by the planned replacement, i.e., weld seams in the steam generator and in the main steam and feedwater piping. Hence, since all the affected areas cannot be tested by Type B or Type C testing, Option B would require that a Type A test be performed prior to startup following the planned steam generator replacement. Type A test measures the containment system overall integrated leakage rate under conditions representing design basis accident containment pressure and system alignment.

However, the affected area of the primary containment boundary is also part of the pressure boundary of an American Society of Mechanical Engineers (ASME) Class 2 component/piping system and, as such, the planned replacement of the steam generators are 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 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 objective of the Type A test is to assure the leak-tight integrity of the area affected by the modification (i.e., the closed system inside Containment formed by the outer shell of the steam generators and the main steam, feedwater, steam generator blowdown, and AFW piping). Although the leak test is in a direction reverse to that of a loss-of-coolant accident environment, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J, Option B with the exception of secondary side access manways. Section 9.2.1, Reference 2 allows reverse testing if justified. Section XI pressure test applies a sealing pressure to the secondary manway due to the inward door swing configuration. Hence, a Type B local leak rate test will be performed for the secondary manways.

For all other affected components, reverse testing is justified since the acceptance criteria for ASME Section XI system pressure testing of welded joints is "zero leakage," and the test pressure for the system pressure test will be approximately 17 times that of a Type A test. Therefore, CCNPP proposes a revision to Technical Specification 5.5.16 to exempt Unit 2 from the requirements of Appendix J, Option B for post-modification integrated leakage rate testing associated with the 2003 steam generator replacement outage. The effect of this amendment request would be to eliminate the post-modification containment integrated leakage rate (Type A) testing required for the modifications to the containment boundary specifically associated with the steam generator replacement.

IV. REFERENCES

1. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995
2. Nuclear Energy Institute 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, dated July 26, 1995, including Errata

ATTACHMENT (2)

DETERMINATION OF SIGNIFICANT HAZARDS

ATTACHMENT (2)

DETERMINATION OF SIGNIFICANT HAZARDS

Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) will be replacing the Combustion Engineering Model 67 original steam generators with replacement steam generators fabricated by Babcock & Wilcox Canada Ltd. during the Unit 1 and Unit 2 refueling and replacement outages in the spring of 2002 and 2003, respectively. In accordance with the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, the ten-year Unit 2 Containment integrated leakage rate (Type A) test was successfully performed during the Spring 2001 refueling outage. The proposed amendment would revise the CCNPP Technical Specification 5.5.16 to exempt CCNPP Unit 2 from the requirements of Appendix J, Option B for post-modification integrated leakage rate testing associated with the 2003 steam generator replacement outage. The Unit 1 ten-year integrated leakage rate test is scheduled to be performed during the Spring 2002 replacement outage; therefore, this proposed amendment does not affect Unit 1 steam generator replacement.

The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The steam generator replacement activities do not affect the containment structure or the actual containment liner. Access for the replacement steam generators as well as removal of the old steam generators will be through the equipment hatch. However, the outer shell of the steam generators, the inside containment portions of the main steam line, the feedwater lines, the auxiliary feedwater lines, and the steam generator blowdown lines are all part of the primary reactor containment boundary that will be impacted by the replacement activities.

   Calvert Cliffs Nuclear Power Plant Technical Specification 5.5.16 states, "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." Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," endorses Nuclear Energy Institute (NEI) 94-01, Revision 0 for methods acceptable to comply with the requirements of Option B. Prior to returning the Containment to operation, NEI 94-01 requires leakage rate testing (Type A testing or local leakage rate testing), following repairs and modification that affect the containment leakage integrity.

   The affected area of the primary containment boundary is also part of the pressure boundary of an American Society of Mechanical Engineers (ASME) Class 2 component/piping system and, as such, the planned replacement of the steam generators are 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 more stringent than the Appendix J, Option B 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 Appendix J, Option B test.

   The objective of the Type A test is to assure the leak-tight integrity of the area affected by the modification. Although the leak test is in a direction reverse to that of the design basis accident environment, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J, Option B with the exception of secondary side access manways. Section 9.2.1, NEI 94-01, Revision 0 allows reverse testing if justified. Section XI pressure test

applies a sealing pressure to the secondary manway due to the inward door swing configuration. Hence, a Type B local leak rate test will be performed for the secondary manways. For all other affected components, reverse testing is justified since the acceptance criteria for ASME Section XI system pressure testing of welded joints is "zero leakage," and the test pressure for the system pressure test will be approximately 17 times that of a Type A test. Hence, the probability or consequences of design basis accidents previously evaluated are unchanged.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

The proposed revision does not involve a physical change to the plant and there are no changes to the operation of the plant that could introduce a new failure mode. As described above in Item 1, the objective of the Appendix J, Option B test is to assure the leak-tight integrity of the area affected by the modification. The ASME Section XI inspection and testing requirements are more stringent than the Appendix J, Option B testing requirements.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment leakage integrated rate testing following replacement of Unit 2 steam generators will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. *Would not involve a significant reduction in the margin of safety.*

As described above in Item 1, the ASME Section XI surface examination, volumetric examination, and system pressure test requirements are more stringent than the Appendix J, Option B 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 Appendix J, Option B test.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators does not involve a significant reduction in the margin of safety.

ATTACHMENT (3)

MARKED-UP TECHNICAL SPECIFICATION PAGE

Page No.

5.0-30

5.5 Programs and Manuals

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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. (Unit 2 is exempted from post-modification integrated leakage rate testing requirement 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.