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Constellation
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Calvert Cliffs
Nuclear Power Plant

A Member of the
Constellation Energy Group

January 31, 2002

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317
License Amendment Request: One-Time Integrated Leakage Rate Test Extension

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) hereby requests an amendment to Renewed Operating License No. DPR-53 to incorporate the changes described below into the Technical Specifications for CCNPP Unit 1.

DESCRIPTION

This proposed one-time change will modify administrative Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" by adding an exception to the Containment Integrated Leakage Rate Test (ILRT) frequency specified in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program" (September 1995) and Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0, July 26, 1995. This change will allow a one-time five-year extension, for a total of 15 years, for the performance of the next Unit 1 ILRT. This request is made on a risk-informed basis as described in Regulatory Guide 1.174. The Combustion Engineering Owners Group (CEOG) has developed the supporting risk-informed information in a Joint Applications Report. This Joint Applications Report was submitted separately by the CEOG under letter CEOG-01-314, dated December 18, 2001, "Transmittal of WCAP-15691 Supplement 1 (Proprietary Information) and WCAP-15715 Supplement 1 (Non-Proprietary), Application of the Joint Application Report to Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2." Further discussion of the change is contained in Attachment (1).

We 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. Because of this modification, and in accordance with the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, we are currently required to complete the next Unit 1 Containment ILRT during the 2002 Refueling Outage. This proposed change will also revise Technical Specification 5.5.16 to exempt Unit 1 from the requirements of Appendix J, Option B for post-modification containment ILRT associated with the 2002 steam

A001

generator replacement outage. The detailed descriptions and the safety analyses for the proposed revision is provided in Attachment (2).

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).

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 Attachments 1 and 2 for a complete discussion). We have also determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase 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 Plant Operations and Safety Review Committee and Offsite Safety Review Committee have reviewed this proposed change and concur that operation with the proposed changes will not result in an undue risk to the health and safety of the public.

SCHEDULE

This change is requested to be approved and issued by April 5, 2002. The requested approval date and implementation period will allow us to optimize refueling outage activities. This request will save critical path time in the upcoming Unit 1 2002 Refueling Outage and defer the ILRT until a subsequent outage.

PRECEDENT

- Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), dated April 17, 2001
- Crystal River Unit 3 – Issuance of Amendment Regarding Containment Leakage Rate Testing Program (TAC No. MB1349), dated August 30, 2001
- Technical Specification Change Request for Waterford 3 SES, Integrated Leakage Rate Testing Interval Extension, dated July 23, 2001

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 31st day of JANUARY, 2002.

WITNESS my Hand and Notarial Seal: Lynn M. Taylor LYNN M. TAYLOR
Notary Public

My Commission Expires: 2/1/2006
Date

CHC/DJM/bjd

Attachments: (1) Proposed Revision to Calvert Cliffs Nuclear Power Plant Technical Specifications One-Time Extension of Unit 1 Integrated Leakage Rate Test Interval Description and Safety Evaluation

(2) Proposed Revision to Calvert Cliffs Nuclear Power Plant Technical Specifications to Support Steam Generator Replacement; Description and Safety Evaluation

Enclosure (1) Plant-Specific Risk Evaluation of ILRT Interval Extension Using Crystal River Plant Methodology

(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 CALVERT CLIFFS NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS ONE-TIME EXTENSION OF UNIT 1 INTEGRATED LEAKAGE RATE TEST INTERVAL DESCRIPTION AND SAFETY EVALUATION

ATTACHMENT (1)

PROPOSED REVISION TO CALVERT CLIFFS NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS ONE-TIME EXTENSION OF UNIT 1 INTEGRATED LEAKAGE RATE TEST INTERVAL DESCRIPTION AND SAFETY EVALUATION

BACKGROUND

Integrated Leakage Rate Tests (ILRTs) have been required of operating nuclear plants to ensure the public health and safety in the event of an accident that would release radioactivity into the Containment. Conservative design and construction practices have led to very few ILRTs exceeding their acceptance criteria. Revisions to 10 CFR Part 50, Appendix J allow individual plants to extend Type A surveillance testing requirements from three in 10 years to at least once per 10 years. The revision to 10 CFR Part 50, Appendix J was based on NUREG-1493, "Performance Based Containment Leak-Test Program," dated September 1995. The NUREG stated that an interval between tests of up to 20 years would contribute an imperceptible increase in risk. The revised Type A test frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than design containment leakage or 1.0 L_a. Calvert Cliffs selected the revised requirements as its testing program. The current Unit 1 ten-year Type A test is due to be performed during the upcoming Unit 1 2002 Refueling Outage currently scheduled for February 2002.

Calvert Cliffs Unit 1 has performed six Type A full-pressure, 50 psig, containment ILRT; a pre-operational Type A test (December 1, 1973) and periodic Type A tests on March 6, 1978, June 22, 1982, May 20, 1985, May 27, 1988 and July 5, 1992. The pre-operational Type A test employed a full pressure (50 psig) and a reduced pressure (25 psig) test. Periodic Type A tests employed both the calculated Mass Point Leakage Rate method and the Total Time Leakage Rate method. Calvert Cliffs currently has a ten-year interval for performing ILRTs. Structural degradation of Containment is a gradual process that occurs due to the effects of pressure, temperature, radiation, chemical, or other factors. Such effects are identified and corrected when the Containment is periodically inspected to verify structural integrity under American Society of Mechanical Engineers (ASME) Section XI, Subsections IWE.

Calvert Cliffs has completed a risk assessment of the proposed one-time Technical Specification change of extending the Containment Type A test interval from once per 10 years to once per 15 years. The risk assessment was performed following the methodology used in Regulatory Guide 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis.

The proposed one-time amendment to the Calvert Cliffs Unit 1 Administrative Technical Specification 5.5.16 would add an exception to the commitment to follow the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." The exception is based on information in Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." The effect of this request will be an extension of the interval since the last ILRT from 10 years to 15 years. We propose to revise Technical Specification 5.5.16 by adding, "except that the next Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007.

Calvert Cliffs is also aware of the discussion between the Nuclear Regulatory Commission (NRC) and NEI concerning a permanent extension of the ILRT interval. The basis for the discussions derives not only from the information in the NUREG, but also from that in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Test Intervals." The one-time change requested here will defer the immediate need for the test and should permit consideration of any agreements reached on the generic change through a revision of NEI 94-01.

SAFETY ANALYSIS

The proposed change to extend the ILRT interval is justified based on a combination of risk-informed analysis and a history of successful Type A tests. The risk aspects of the justification have been prepared by the Combustion Engineering Owners Group (CEOG) and are presented in a Joint Applications Report, WCAP-15691, Revision 01. That report has been transmitted for NRC review separately from this submittal. A brief system description and a history of Calvert Cliffs Unit 1 Type A testing is also provided in the report (see WCAP-15691, Supplement 1, Appendix B, Section B1.2).

The Joint Applications Report provides the risk-informed supporting analyses to demonstrate that the increase in risk of extending the ILRT interval from 10 to 15 years is insignificant. That analysis, done in accordance with Regulatory Guide 1.174, shows that the increase in total plant risk due to the extended ILRT interval is well under one-half of one percent. The delta-Large Early Release Frequency (LERF) is only 1.2E-8/year for an increase from 10 years to 15 years. Note that the Joint Applications Report also demonstrates that, from a risk perspective, an extension in the interval out to 20 years has an insignificant impact on risk. This is consistent with the findings of NUREG-1493. However, this submittal requests only a one-time extension from 10 years to 15 years.

The risk assessment documented in the Joint Applications Report demonstrates:

1. The risk of extending the ILRT interval for Type A tests from its current interval of 10 years to 15 years was evaluated for potential public exposure impact (as measured in person-rem/year) as described in the Joint Applications Report. The risk assessment predicts a slight increase in risk when compared to that estimated from current requirements. For the change from a 10-year test interval to a 15-year test interval, the increase in the risk (person-rem/year within 50 miles) was found to be 0.07 percent. Note that the cumulative increase in risk, given the change from the original 3 in 10-year test interval to a 15-year test interval, was found to be 0.16 percent. These results agree with the conclusion in NUREG-1493. NUREG-1493 concluded this represents an imperceptible increase in risk. Therefore, the increase in the risk for the proposed change is considered small.

2. Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in the risk guidelines as increases in Core Damage Frequency (CDF) less than 1E-6 per reactor year and increased in LERF less than 1E-7 per reactor year. Since the Type A test does not impact CDF, the relevant criterion in evaluating this proposed change is LERF. The increase in LERF resulting from a change in the Type A test frequency from the current one in 10 years to one in 15 years is estimated to be 1.2E-8/year. The cumulative increase in LERF resulting from a change in the Type A test interval from the original three in 10 years to one in 15 years is estimated to be 2.8E-8/year. Increasing the Type A test interval to 15 years is considered to be a very small change in LERF.

3. Regulatory Guide 1.174 also encourages the use of risk analysis techniques to help ensure 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 LERF probability was estimated to be 0.08 percent for the proposed change and 0.18 percent for the cumulative change of going from a test interval of three in 10 years to one in 15 years. We conclude that the very small impact on the LERF probability demonstrates that consistency with the defense-in-depth philosophy is maintained for the proposed change.

In the Joint Applications Report, undetected containment leakage due to Type A ILRT extension is considered. The amount of leakage is based on a statistical analysis of historical leakage that could only be detected by Type A testing. This leakage is binned into the Class 3A and Class 3B categories. The Joint Applications Report assessment shows that even with extension, the increase in risk is insignificant.

As part of the Individual Plant Examination (IPE), the containment response to severe accident analysis was determined. Seven key containment failure modes exist (Reference 1, Table 4.4.1). Each of these failure modes has a different likelihood of failing as a function of temperature and pressure. These containment failure likelihoods combine to form a composite failure likelihood (Reference 1, Figure 4.4.1-A).

The containment response developed in the IPE is still used in the current Calvert Cliffs Nuclear Power Plant risk calculations.

LERF Modeling

The LERF Model used in the ILRT analysis includes additional release contributions that are currently not included in the Calvert Cliffs PRA (CCPRA). These were added to ensure that the total LERF contribution is bounded. They also resulted in increasing the calculated ILRT LERF to above 1E-5. The LERF modeling used for the CCPRA and for this ILRT analysis is discussed below to properly characterize the calculated LERF.

The CCPRA Level 2 modeling, including the calculation of LERF, represents a slight improvement over the Individual Plant Examination (IPE and Integrated Plant Evaluation for External Events) modeling and uses many of the same assumptions and documentation as the IPE. The core damage sequences are grouped into plant damage states based on equipment availability, the status of the Reactor Coolant System, the status of the steam generators, and the status of Containment. Similar plant damage states are further grouped into key plant damage states. The key plant damage states are evaluated using a containment event tree to determine the likelihood of release. Certain release states are considered to be large early releases.

The CCPRA release categories "Early Large Containment Failure" and "Large Containment Bypass" are used as surrogates for LERF. The ILRT analysis includes the additional contribution from "Small Containment Bypasses" and 50 percent of the "Early Small Containment Failures." Both analyses include the LERF contribution from internal events, fires, seismic events, and high winds. The additional ILRT categories result in an ILRT LERF that is a factor of two higher.

The annual CCPRA LERF for Unit 1 is 7.6E-6 with the following breakdown.

Evaluation	Contribution
Fire	55%
Seismic	24%
Internal Events	20.3%
Wind	0.7%

A similar breakdown for the ILRT LERF is not available.

The differences between the two analyses are rooted in the uncertainty associated with the size of large early releases. It is possible that some small early releases and small bypasses could be considered large depending on the basis for a large release. The ILRT LERF analysis estimated this by including all small bypasses and 50 percent of the early small containment failures in the determination of LERF.

In the CCPRA, one of the dominant contributors to the small bypass is Steam Generator Tube Rupture (SGTR) (7E-6). The success criteria for SGTR is that secondary side heat removal is available and safety injection is available to provide inventory control. Currently, we do not credit steam generator isolation as an alternative to having safety injection available. During our recent peer review, the reviewers classified this as too conservative. The vast majority of the SGTR sequences have auxiliary feedwater available with no safety injection (over 90 percent). These sequences can be recovered by isolating the steam generators as an alternate to safety injection. Therefore, the ILRT LERF is conservative in that these potentially isolable SGTRs are included. Future updates of the CCPRA will address this issue. If these sequences were removed from the ILRT LERF, the value would be below 1E-5.

There are other areas of uncertainty associated with the CCPRA LERF analysis. These include:

- The impact of upgrading the containment event tree thermal-hydraulic code from MAAP 3.0b to MAAP 4.0.
- The impact of upgrading the containment release related top events. These top events have only received minor updates since the IPE analysis.
- The need for a LERF truncation sensitivity analysis. Although a CDF truncation sensitivity analysis has been done, a similar analysis has not been done for LERF.
- The incorporation of the updated fire PRA results. Preliminary results have shown a reduced risk contribution from fire.

The overall impact of incorporating these changes is not known. However, it is believed that these and other modeling refinements would more likely reduce rather than increase, the calculated LERF.

Enclosure (1) contains a sensitivity evaluation comparing the CEOG Joint Applications Report Methodology with the methodology that was approved for the Indian Point 3 application.

Inspections

Containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the ASME Code Boiler and Pressure Vessel Code, Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulation 10 CFR 50.55a(b)(2)(ix)(E) requires licensees to conduct a general visual inspection of the Containment in accordance with ASME XI during each inspection period. Also, inspections required by the Maintenance Rule (10 CFR 50.65) may identify containment degradation that could affect leak tightness. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

The Calvert Cliffs ASME Section XI, IWE/IWL Inservice Inspection Program Plan was developed in accordance with the final rule amending 10 CFR 50.55a, effective September 9, 1996 which referenced the 1992 Edition through 1992 Addenda of ASME Section XI, Subsections IWE and IWL.

Provided below is a definition of the examination methods which will be performed to satisfy the ASME Section XI requirements.

Visual Examination Method

For Subsection IWE, visual examinations (VT) will be performed in accordance with IWA-2210, IWE-3510, and IWE-3512 of ASME Section XI. IWA-2210 and IWE-3510 define types and extent of VT examinations as follows:

- General VT examinations are conducted either directly or remotely with sufficient illumination (natural or artificial) and resolution suitable for the local environment to assess the general condition of the containment surfaces from permanent vantage points, e.g., floors, roofs, platforms, ladders, etc. The objective of this examination is to detect evidence of damage, deterioration, etc., that may affect either containment structural integrity or leak tightness. The general VT examinations are required on all accessible surface areas of the containment liner once every inspection period.
- VT-1 examinations are conducted to determine the condition of the part, component, or surface examined. The examination shall determine conditions such as cracks, wear, corrosion, erosion, or physical damage on the surfaces of the parts or components. VT-1 examinations are required on all Category E-C, Surfaces Requiring Augmented Examination and Category E-G, Pressure Retaining Bolting. All Category E-G Bolted Connections are required to be examined once during each inspection interval. Calvert Cliffs Nuclear Power Plant Unit 1 currently has one area designated in accordance with IWE-1242 as a Category E-C area, this is the area of the liner immediately below the moisture barrier at the juncture of the containment floor and the liner plate wall. Previous examinations have found corrosion in this area that did not violate the minimum wall required. These areas will need to be examined every examination period until it is determined there has been no change in the condition for three consecutive periods.
- VT-3 examinations are conducted to determine the general mechanical and structural condition of components such as the presence of loose parts, debris, or abnormal corrosion products, wear, erosion, corrosion, and the loss of integrity at bolted or welded connections as defined in IWA-2213. VT-3 examinations are required on Category E-A; Accessible Containment Surfaces one per inspection interval in accordance with Table IWE-2500-1. VT-3 examinations are also performed prior to the removal of containment coatings and following re-application of the coatings.

For Subsection IWL, VT examinations will be performed in accordance with ASME Section XI, IWA-2210 and IWL-2300. The three types of VT examinations are as follows.

- VT-1C examinations are conducted to determine concrete deterioration and distress for suspect areas detected by VT-3C, and the condition (e.g., cracks, wear, or corrosion) of tendon anchorage or strands.

- VT-3C examinations are conducted to determine the general structural condition of concrete surfaces of Containment by identifying areas of concrete deterioration and distress, such as defined in American Concrete Institute 201.1, R-68. VT-3C examinations are performed on all non-exempt concrete containment surfaces every five years in accordance with IWL-2410.
- VT-1 examinations are conducted to determine the condition of the part, component, or surface examined. The examination shall determine conditions such as cracks, wear, corrosion, erosion, or physical damage on the surfaces of the parts or components.

Volumetric Examination Method

For Subsection IWE, a volumetric examination is performed to detect material thickness. Ultrasonic (UT) examinations will be conducted as required in accordance with IWE-2500(c)(4). The UT examinations will be performed utilizing either manual or mechanized UT techniques. The UT examinations are performed in conjunction with the VT-1 examinations described above for the Category E-C areas.

In addition to these examinations the Concrete Post Tensioning System is subject to the testing required by ASME Section XI in accordance with IWL-2520.

The NRC approved Inservice Inspection Relief Request E1 for seals and gaskets on November 16, 1998. As stated in this relief request, the alternate examinations of Appendix J Type B testing 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 Ten-Year Inspection Interval.

Information Notice 92-20, "Inadequate Local Leak Rate Testing," discussed the inadequate local leak rate testing of two-ply stainless steel bellows. Calvert Cliffs has no such bellows that act as part of the containment boundary.

DETERMINATION OF SIGNIFICANT HAZARDS

The proposed amendment to the Calvert Cliffs Unit 1 Administrative Technical Specification 5.5.16 would add a one-time exception to the commitment to follow the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." The exception is based on information in Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The effect of this request will be an extension of the interval since the last Integrated Leakage Rate Test (ILRT) from 10 years to 15 years.

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

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

   This proposed one-time extension of the Type A test interval does not increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493 has found that, generically, fewer than three percent of the potential containment leak paths are not identified by Type B and C testing. Calvert Cliffs, through testing and containment inspections, also provides a high degree of assurance that the Containment will not degrade in a manner detectable only by a Type A test. Inspections required by the Maintenance Rule (10 CFR 50.65) and by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leak tightness.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   This proposed one-time extension to the interval for the Type A test does not involve any design or operational changes that could lead to a new or different kind of accident from any accident previously evaluated. The test itself is not changing and will be performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

   Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The generic study of the increase in the Type A test interval, NUREG-1493, concluded there is an imperceptible increase in the plant risk associated with extending the test interval out to 20 years. Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detect 97 percent of potential leakage paths. For the requested change in the Calvert Cliffs Integrated Leakage Rate Test interval, it was determined that the risk contribution of leakage will increase 0.07 percent (based on change in offsite dose). This change is considered very small and does not represent a significant reduction in the margin of safety.

   Therefore, this change does not involve a significant reduction in the margin of safety.

REFERENCE

1. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated December 30, 1993, "Summary Report of Individual Plant Examination Results (Generic Letter 88-20) (TAC Nos. M74392 & M74393)

ENCLOSURE (1)

Plant-Specific Risk Evaluation of ILRT Interval Extension

Using Crystal River Plant Methodology

Calvert Cliffs Nuclear Power Plant, Inc.
January 31, 2002

Sensitivity Evaluation Comparing the Combustion Engineering Owners Group Joint Applications Report Methodology with an Alternate Previously Approved Methodology

The Calvert Cliffs Nuclear Power Plant (CCNPP) submittal references the Combustion Engineering Owners Group (CEOG) Joint Applications Report (JAR) as the supporting technical justification for the request of a one-time extension of the Integrated Leakage Rate Test (ILRT) interval to 15 years.

The purpose of this enclosure is to present information about the plant-specific analysis using a methodology similar to that already approved for the Crystal River Unit 3 (CR-3) application. Note that CCNPP believes the methodology applied in the CEOG JAR to be reasonable and consistent with good practice in risk-informed evaluations. The results of the CEOG evaluation, which represents the use of a best-estimate approach to establish the probability of the small isolation failures of interest, demonstrates an even better risk justification of the request. The previously approved CR-3 methodology utilizes a 95th percentile estimate of the probability of the small isolation events and the results reflect a somewhat greater impact of the change on overall risk. Other differences between the methodologies (CEOG JAR and CR-3) will be described in the body of the evaluation below. The change is demonstrated to be risk insignificant in both methodologies.

Both of the methodologies followed the same general approach to the evaluation of the risk of the interval extension. There were differences in the approaches in the assumptions and in the development of a probability estimate for the release class 3 events. The methodologies:

- both utilize the Electric Power Research Institute (EPRI) TR-104285 release classes to categorize the various containment failure scenarios.
- both establish the plant-specific frequencies for each EPRI release class.
- both define estimated leakage for each release class.
- both quantify the risk for each release class by multiplying the class frequency times the assumed leakage.
- both evaluated a baseline case (three tests in 10 years), a current case (one test in 10 years), and the proposed case (one test in 15 years).

Table 1 summarizes the treatment of each of the EPRI Release Classes and provides a summary of some of the differences between the CEOG JAR and the CR-3 methodologies.

Table 1
EPRI Release Class Definitions

Release Class	Description	CR-3 Submittal	CEOG JAR
1	No containment failure	Frequency reduced as Class 3 increases; leakage magnitude increases to 2 L_a	Frequency reduced with Class 3 increase; considered leakage of L_a
2	Large isolation failures	No change from baseline consequence measures; considered leakage of 35 L_a	No change from baseline consequence measures; considered leakage of 500 L_a
3	Isolation failures	3a: small leaks, 10 L_a, non-Large Early Release Frequency (LERF) 3b: large leaks, 35 L_a, LERF Probability derived using 95th %-ile χ^2 distribution of NUREG-1493 data	3a: small leaks, 25 L_a, non-LERF 3b: large leaks, 500 L_a, LERF Probability derived using log-normal distribution of NUREG-1493 data
4,5	Other small isolation failures [local leak rate test (LLRT)]	No change from baseline consequence measures; not analyzed	No change from baseline consequence measures; not analyzed
6	Other isolation failures	No change from baseline consequence measures; considered leakage of 35 L_a	No change from baseline consequence measures; considered leakage of 175 L_a
7	Induced failures	No change from baseline consequence measures; considered leakage of 100 L_a	No change from baseline consequence measures; considered leakage of 1400 L_a
8	Bypass	Characterized by steam generator tube rupture (SGTR) scenario – not impacted by ILRT extension	Characterized by SGTR and inter system loss-of-coolant accident – not impacted by ILRT extension

Evaluation of Baseline ILRT Interval

The plant-specific evaluation of risk for the baseline case ILRT interval for Calvert Cliffs is presented in Table 2. The release frequencies for the Class 2, 6, 7, and 8 bins are taken from the CEOG JAR, which had compiled this data based on the Calvert Cliffs Probabilistic Safety Analysis. As noted in Table 1, the risk associated with the Class 4 and 5 bins, is not impacted by the ILRT interval and is not analyzed here. The release frequencies for the Class 3a and 3b bins are determined based on the previously approved methodology (see next paragraph). The release frequency for Class 1 is the value of core damage frequency (CDF) reduced by the frequencies of the Class 3a and 3b scenarios. (Note – the CEOG JAR had utilized a value of CDF representative of sequences in which the Containment remains intact. This value was approximately 52 percent of the total CDF. The previously approved methodology used total CDF. Total CDF will be used in this plant-specific evaluation.)

The Class 3a and 3b frequencies in the previously approved methodology were determined based on a 95th percentile χ^2 distribution of the NUREG-1493 data. For the baseline ILRT interval (three tests in 10 years), this resulted in a frequency for Class 3a of 0.064 (Reference 1) times CDF and a frequency for Class 3b of 0.021 (Reference 2) times CDF. These frequencies are used in the Calvert Cliffs evaluation presented in Table 2. Note the total CDF for Calvert Cliffs is 1.1E-04 per year and the intact containment release frequency is 4.81E-05 per year based on the current plant risk model.

Table 2
Calvert Cliffs Risk Evaluation of Baseline ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	FREQ(intact)-FREQ(3a)-FREQ(3b)-FREQ(6) = 3.73E-05 (Reference 4)	L_a = 9.79E+05 (Reference 3)	36.56
2	4.97E-08	35 L_a = 3.43E+07	1.70
3a	0.064 x CDF = 7.04E-06	10 L_a = 9.79E+06	68.92
3b	0.021 x CDF = 2.31E-06	35 L_a = 3.43E+07	79.15
6	1.41E-06	35 L_a = 3.43E+07	48.31
7	5.43E-05	100 L_a = 9.79E+07	5315.97
8	6.47E-06	5.47E+08 (Reference 3)	3539.09
		Total Risk	9089.71

In the CEOG JAR, a risk contribution of the intact containment sequences (i.e., Classes 1, 3a, and 3b) was determined. Using the previously approved methodology, the risk contribution due to the ILRT Type A testing was considered to be due to the Class 3a and 3b scenarios. From Table 2, it can be seen that the risk contribution associated with the ILRT interval considering Classes 3a and 3b is:

$$\begin{aligned}\% \text{ Risk} &= [(\text{Risk}_{\text{Class 3a}} + \text{Risk}_{\text{Class 3b}}) / \text{Total Risk}] \times 100 \\ &= [(68.92 + 79.15) / 9089.71] \times 100 \\ &= 1.63\%\end{aligned}$$

In the CEOG JAR, it was also assumed that the Class 2, 3b, 6, 8, and a fraction of the Class 7 are considered part of the LERF. The fraction of Class 7 considered a LERF is half of the early small containment failure scenarios and all the large early containment failure scenarios not already considered in other classes. The previously approved methodology focused only on the Class 3b scenario, which is the only one affected by the consideration of the ILRT interval. As the parameter of concern in the evaluation is ΔLERF, and because Class 3b is the only class affected by the interval extension, ΔLERF is compared on a consistent basis in both methodologies. Thus, for this evaluation the baseline LERF is the Class 3b frequency, or 2.31E-06 per year.

Risk Evaluation of the Current ILRT Interval (1 in 10 years)

This evaluation of the 'once in 10 years' interval will be performed using the same approach as taken above for the baseline case. The frequencies for all release classes, except Class 1, 3a, and 3b, are unaffected by the change in the interval and remain as in Table 2. And the releases for all of the classes are the same as those shown in Table 2 for the baseline case.

The increased probability of not detecting excessive leakage in a Type A test directly impacts the frequencies of the Class 3 events. Based on the previously approved methodology, the Class 3a and 3b frequencies are determined by simply multiplying the baseline frequency by a factor of 1.1. With this change in the Class 3 frequencies, the Class 1 frequency is also adjusted to preserve the total CDF. The evaluation of the current interval is presented in Table 3.

Table 3
Calvert Cliffs Risk Evaluation of Current ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	FREQ(intact)-FREQ(3a)-FREQ(3b)-FREQ(6) = 3.64E-05 (Reference 4)	L_a = 9.79E+05 (Reference 3)	35.64
2	4.97E-08	35 L_a = 3.43E+07	1.70
3a	1.1 x 0.064 x CDF = 7.74E-06	10 L_a = 9.79E+06	75.81
3b	1.1 x 0.021 x CDF = 2.54E-06	35 L_a = 3.43E+07	87.16
6	1.41E-06	35 L_a = 3.43E+07	48.31
7	5.43E-05	100 L_a = 9.79E+07	5315.97
8	6.47E-06	5.47E+08 (Reference 3)	3539.09
		Total Risk	9103.68

As was noted above for the baseline evaluation:

- the risk contribution due to the Type A test interval is [(75.81 + 87.16) / 9103.68] x 100, or 1.79%.
- the LERF for the current interval evaluation is the Class 3b frequency, or 2.50E-06 per year.

Risk Evaluation of the Proposed ILRT Interval (1 in 15 years, one-time)

This evaluation of the 'once in 15 years' interval will be performed using the same approach as taken above for the baseline case. The frequencies for all release classes, except Class 1, 3a, and 3b, are unaffected by the change in the interval and remain as in Table 2. And the releases for all of the classes are the same as those shown in Table 2 for the baseline case.

The increased probability of not detecting excessive leakage in a Type A test directly impacts the frequencies of the Class 3 events. Based on the previously approved methodology, the Class 3a and 3b frequencies are determined by simply multiplying the baseline frequency by a factor of 1.15. With this change in the Class 3 frequencies, the Class 1 frequency is also adjusted to preserve the total CDF. The evaluation of the current interval is presented in Table 4.

Table 4
Calvert Cliffs Risk Evaluation of Proposed ILRT Interval

Class	Frequency (per reactor-year)	Release (person-rem)	Risk (person-rem/year)
1	CDF- freq(3a)-freq(3b) = 3.59E-05	L_a = 9.79E+05	35.18
2	4.97E-08	35 L_a = 3.43E+07	1.70
3a	1.15 x 0.064 x CDF = 8.10E-06	10 L_a = 9.79E+06	79.26
3b	1.15 x 0.021 x CDF = 2.66E-06	35 L_a = 3.43E+07	91.12
6	1.41E-06	35 L_a = 3.43E+07	48.31
7	5.43E-05	100 L_a = 9.79E+07	5315.97
8	6.47E-06	5.47E+08	3539.09
		Total Risk	9110.63

As was noted above for the baseline evaluation:

- the risk contribution due to the Type A test interval is [(79.26 + 91.12) / 9110.63] x 100, or 1.87%.
- the LERF for the current interval evaluation is the Class 3b frequency, or 2.70E-06 per year.

Conditional Containment Failure Probability

Another parameter of interest in evaluating the risk impact of a change to the ILRT interval is the conditional containment failure probability (CCFP). In the CEOG JAR methodology, ΔLERF was considered to be directly related to ΔCCFP. The results using that approach were a ΔCCFP of 0.08 percent due to the proposed interval compared to the current interval, and 0.20 percent due to the change to the proposed interval compared to the baseline case. In the previously approved methodology that was used in the plant-specific evaluation developed in this submittal, CCFP was defined as:

CCFP = 1 – (frequency of no containment failure sequences / CDF)

Further, the sequences representing no containment failure were considered to be the Class 1 and 3a events. Thus, using this approach and the information from Tables 2, 3, and 4, the ΔCCFP between the current ILRT interval and the proposed ILRT interval may be derived by:

$$\begin{aligned}\Delta CCFP_{c\ to\ p} &= \{[\text{freq (Cl1)} + \text{freq (Cl3a)}]_c - [\text{freq (Cl1)} + \text{freq (Cl3a)}]_p\} / \text{CDF} \\ &= \{[3.64E\text{-}05 + 7.74E\text{-}06] - [3.59E\text{-}05 + 8.10E\text{-}06]\} / 1.10E\text{-}04 \\ &= 0.0013, \text{ or } 0.13\%\end{aligned}$$

Similarly, the impact of the proposed ILRT interval compared with the baseline ILRT interval is given by:

$$\Delta CCFP_{b\ to\ p} = \{[freq\ (Cl1) + freq\ (Cl3a)]_b - [freq\ (Cl1) + freq\ (Cl3a)]_p\} / CDF$$
$$= \{[3.73E\text{-}05 + 7.04E\text{-}06] - [3.59E\text{-}05 + 8.10E\text{-}06]\} / 1.10E\text{-}04$$
$$= 0.0031, \text{ or } 0.31\%$$

Summary

A summary of the risk evaluation of the ILRT interval changes using the previously approved methodology is presented in Table 5.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 1E-06/year and increases in LERF below 1E-07/year. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF involves determining the impact of the ILRT interval on the leakage probability.

Table 5
Summary of Results of ILRT Interval
Risk Evaluation (Using Previously Approved Approach)

ILRT Interval	ILRT Risk Contribution	LERF (per year)	ΔLERF from baseline (per year)	ΔLERF from current (per year)
baseline (3 in 10 years)	1.63%	2.31E-06	—	—
current (1 in 10 years)	1.79%	2.54E-06	2.30E-07	—
proposed (1 in 15 years)	1.87%	2.66E-06	3.50E-07	1.20E-07

For comparison purposes, the evaluation results from the CEOG JAR, derived using differences in assumptions and methodology, are presented in Table 6

Table 6
Summary of Results of ILRT Interval
Risk Evaluation (using CEOG JAR approach)

ILRT Interval	ILRT Risk Contribution	LERF*	ΔLERF from baseline	ΔLERF from current
baseline (3 in 10 years)	0.11%	1.533E-05	—	—
current (1 in 10 years)	0.20%	1.534E-05	1.616E-08	—
proposed (1 in 15 years)	0.26%	1.536E-05	2.828E-08	1.212E-8

* Note that in performing the CEOG baseline frequency calculations, the LERF value is larger than the CCNPP LERF calculation. The CEOG LERF includes half of the small containment failure frequency

and all of the small containment by-pass frequency. The CCNPP LERF only considers the large containment release states outlined in the Individual Plant Examination submittal. The current CCNPP LERF is 7.8E-6 (for Internal and External Events). The delta increase associated with the ILRT Type "A" extension is 1.2E-8. The combination of these still results in a CCNPP LERF less than 1E-5.

Conclusion

The risk associated with extending the ILRT interval is quantifiable. Calvert Cliffs Nuclear Power Plant has utilized two alternate methodologies to quantify the risk and evaluate the proposed change in the ILRT interval to 15 years. Both methodologies demonstrate the risk associated with the extension of the interval is small. On this basis, CCNPP requests approval of a one-time extension of the Calvert Cliffs ILRT interval to 15 years.

References

1. FPC Calculation F-01-0001, Revision 2, Evaluation of Risk Significance of ILRT Extension, page 12
2. FPC Calculation F-01-0001, Revision 2, Evaluation of Risk Significance of ILRT Extension, page 11
3. WCAP-15691 Revision 1, Appendix B, Table B2-6
4. FPC Calculation F-01-0001, Revision 2, Evaluation of Risk Significance of ILRT Extension, page 13

ATTACHMENT (2)

PROPOSED REVISION TO CALVERT CLIFFS NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

ATTACHMENT (2)

PROPOSED REVISION TO CALVERT CLIFFS NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

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 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 replacement steam generators will occupy the same physical envelope as the original steam generators. There are no changes to interfaces with the Reactor Coolant, Main Feedwater, or Main Steam Systems, or to major component supports or piping supports.

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 post-modification containment integrated leakage rate testing following replacement of Unit 1 steam generators.

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.

As described in Attachment (1), Calvert Cliffs is requesting an extension of the integrated leakage rate test (ILRT) interval for Unit 1. In addition to the extension, the proposed amendment would exempt CCNPP Unit 1 from the requirements of Appendix J, Option B for a post-modification ILRT associated with the 2002 steam generator replacement outage. A similar request to exempt CCNPP Unit2 from the requirements of Appendix J, Option B for a post-modification ILRT associated with the 2003 steam generator replacements was submitted in November, 2001 (see Reference 3).

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

SAFETY ANALYSIS

The CCNPP Unit 1 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 1 steam generators includes the following activities:

- Cutting and removing the main steam, feedwater, and other necessary 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 1 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 ILRT (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 1 from the requirements of Appendix J, Option B for post-modification ILRT associated with the 2002 steam generator replacement outage. The effect of this amendment request would be to eliminate the post-modification containment ILRT (Type A) required for the modifications to the containment boundary specifically associated with the steam generator replacement.

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. The proposed amendment would revise the CCNPP Technical Specification 5.5.16 to exempt CCNPP Unit 1 from the requirements of Appendix J, Option B for post-modification integrated leakage rate testing associated with the 2002 steam generator replacement outage.

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 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 1 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 integrated leakage rate testing following replacement of

Unit 1 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 1 steam generators does not involve a significant reduction in the margin of safety.

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
3. Letter from Mr. C. H. Cruse (CCNPP) to NRC Document Control Desk, dated November 19, 2001, "License Amendment Request: Revision to the Containment Leakage Rate Testing Program Technical Specification to Support Steam Generator Replacement"

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, As modified by the following exceptions:

INSERT A

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.

INSERT A

1. 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.
2. Unit 1 is exempted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.