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REF 10 CFR 50.90

January 28, 2015

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Comanche Peak Nuclear Power Plant
Docket Nos. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST 14-002
EXTENSION OF CONTAINMENT LEAKAGE RATE TESTING PROGRAM

Dear Sir or Madam:

Pursuant to 10CFR50.90, Luminant Generation Company LLC (Luminant Power) hereby requests an amendment to the CPNPP Unit 1 Operating License (NPF-87) and CPNPP Unit 2 Operating License (NPF-89) by incorporating the attached changes into the CPNPP Unit 1 and 2 Technical Specifications (TSs). Proposed change LAR 14-002 is a request to revise Technical Specifications (TS) 5.5.16, "Containment Leakage Rate Testing Program," for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2. This change request applies to both Units.

The proposed change to the Technical Specifications (TS) contained herein would revise CPNPP TS 5.5.16, by replacing the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as the implementation documents used by CPNPP to implement the Unit 1 and Unit 2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

The proposed change would allow an increase in the Integrated Leak Rate Test (ILRT) test interval from its current 10 year frequency to a maximum of 15 years and the extension of the containment isolation valves leakage test (Type C tests) from its current 60 month frequency to 75 months in accordance with NEI 94-01 Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A. The proposed change would also delete the listing of one time exceptions previously granted to Integrated Leak Rate Test (ILRT) test frequencies.

Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, Luminant Power's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected Unit 1 and Unit 2 Technical

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Specification (TS) page marked-up to reflect the proposed change. Attachment 3 provides the affected Unit 1 and Unit 2 Technical Specification Bases (TSB) page marked-up to reflect the proposed change for information only. Attachment 4 provides the retyped TS page which incorporates the requested change. Attachment 5 provides the retyped TSB page for information only. Attachment 6 provides the PRA Evaluation of the Permanent ILRT Extension Risk Impact Assessment. Attachment 7 provides the CPNPP PRA Model technical adequacy.

Luminant Power requests approval of the proposed license amendment by December 2, 2015, to support the CPNPP Unit 1 Spring 2016 (1RF18) refueling outage and not have to conduct a Unit 1 Containment ILRT. As concluded in Attachment 6, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the CPNPP Unit 1 and Unit 2 risk profile. The proposed license amendment will be implemented within 120 days of issuance of the license amendment.

In accordance with 10 CFR 50.91(b), Luminant Power is providing the State of Texas with a copy of this proposed amendment.

If you have any questions regarding this request, please contact Jack Hicks at 254-897-6725 or jack.hicks@luminant.com.

I state under penalty of perjury that the foregoing is true and correct.

Executed on January 28, 2015.

Sincerely,

Luminant Generation Company LLC

Rafael Flores

By: Fred W. Madden
Director, External Affairs

Attachments:
1. Description and Assessment
2. Proposed Technical Specifications Change (Mark-Up)
3. Proposed Technical Specification Bases Change (Mark-Up)
4. Retyped Technical Specifications Page
5. Retyped Technical Specification Bases Page
6. CPNPP PRA Evaluation Permanent ILRT Extension Risk Assessment
7. CPNPP PRA Technical Adequacy

c - Mark L. Dapas, Region IV
Balwant K. Singal, NRR
Resident Inspectors, CPNPP

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ATTACHMENT 1 to TXX-15001
DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

By this letter, Luminant Generation Company LLC (Luminant) requests an amendment to the CPNPP Unit 1 Operating License (NPF-87) and CPNPP Unit 2 Operating License (NPF- 89) by incorporating the attached change into the CPNPP Unit 1 and 2 Technical Specifications. Proposed change LAR 14-002 is a request to revise Technical Specifications (TS) 5.5.16, "Containment Leakage Rate Testing Program," for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2. Luminant requests approval of the proposed License Amendment by December 2, 2015, to be implemented within 120 days of the issuance of the license amendment.

The proposed change to the Technical Specifications (TS) contained herein would revise CPNPP TS 5.5.16, by replacing the reference to Regulatory Guide (RG) 1.163 (Reference 1) with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 3-A (Reference 2) and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 3), dated October 2008, as the implementation documents used by CPNPP to implement the Unit 1 and Unit 2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

The proposed change would allow an increase in the Integrated Leak Rate Test (ILRT) test interval from its current 10 year frequency to a maximum of 15 years and the extension of the containment isolation valves leakage test (Type C tests) from its current 60 month frequency to 75 months in accordance with NEI 94-01 Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A. The proposed change would also delete the listing of one-time exceptions in TS 5.5.16a.3 previously granted to Integrated Leak Rate Test (ILRT) test frequencies.

The Bases for CPNPP Technical Specification SR 3.6.3 will be updated as a result of this License Amendment Request.

CPNPP Final Safety Analysis Report Chapter 6 will be updated as a result of this License Amendment Request.

2.0 PROPOSED CHANGE

CPNPP Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" currently states, in part:

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995" as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

3. NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)."

The proposed change to CPNPP Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" will delete paragraph a.3 of TS 5.5.16 and replace the reference to Regulatory Guide 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 3-A and 2-A. The proposed change will revise Technical Specification 5.5.16 to state, in part:

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

   Sub-paragraphs a.1 and a.2 of TS 5.5.16 would be retained in their entirety.

A markup of the proposed change to Technical Specification 5.5.16 is provided in Attachment 2.

This proposed change is requested to extend the performance of the next Unit 1 ILRT from Spring 2016 refueling outage to a subsequent refueling outage no later than Spring 2022 when it can be performed in a refueling outage that involves fewer conflicts with other planned activities and without extending the refueling outage duration. This proposed amendment would also extend the performance of the next Unit 2 ILRT to be performed no later than Fall 2027.

The TS Bases for SR 3.6.3 is revised for consistency with the adoption of NEI 94-01 Revision 3-A and the deletion of RG 1.163.

A markup of the proposed change to Technical Specification Bases for SR 3.6.3 is provided in Attachment 3.

The retyped Technical Specification 5.5.16 is provided in Attachment 4.

The retyped Technical Specification Bases for SR 3.6.3 is provided in Attachment 5.

Attachment 6 contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174 (Reference 4) and NRC Regulatory Guide

1.200, Revision 2 (Reference 5). The risk assessment concluded that the increase in risk as a result of this proposed change is small and is well within established guidelines.

Attachment 7 provides a description of the CPNPP PRA Technical Adequacy.

3.0 BACKGROUND

3.1 Justification for the Technical Specification Change

3.1.1 Chronology of Testing Requirements of 10 CFR 50 Appendix J

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. 10 CFR 50, Appendix J also ensures that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50 Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, Regulatory Guide (RG) 1.163 was issued. The RG endorsed Nuclear Energy Institute (NEI) 94-01, Revision 0, (Reference 6) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 7) and Electric Power Research Institute (EPRI) TR-104285 (Reference 8) both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months were considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but that this "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A, (Reference 3) was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC Safety Evaluation Report (SER) on NEI 94-01. The NRC SER was included in the front matter of this NEI report. NEI 94-01, Revision 2-A, includes provisions for extending Type A ILRT intervals to up to fifteen years and incorporates the regulatory positions stated in Regulatory Guide 1.163 (September 1995). It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

In 2012, NEI 94-01, Revision 3-A, was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J and includes provisions for extending Type A ILRT intervals to up to fifteen years. NEI 94-01 has been endorsed by Regulatory Guide 1.163 and NRC SERs of June 25, 2008 (Reference 9) and June 8, 2012 (Reference 10) as an acceptable methodology for complying with the provisions of Option B to 10 CFR Part 50. The regulatory positions stated in Regulatory Guide 1.163 as modified by NRC SERs of June 25, 2008 and June 8, 2012 are incorporated in this document. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic as-found tests where the results of each test are within a licensee's allowable administrative limits. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2.

The NRC has provided the following concerning the use of grace in the deferral of ILRTs past the 15 year interval in NEI 94-01, Revision 2-A, NRC SER Section 3.1.1.2:

> "As noted above, Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists."

NEI 94-01, Revision 3-A, Section 10.1 concerning the use of grace in the deferral of Type B and Type C LLRTs past intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing, states:

> "Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing given in this section may be extended by up to 25 percent of the test interval, not to exceed nine months.
>
> Notes: For routine scheduling of tests at intervals over 60 months, refer to the additional requirements of Section 11.3.2.
>
> Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. This provision (nine month extension) does not apply to valves that are restricted and/or limited to 30 month intervals in Section 10.2 (such as BWR MSIVs) or to valves held to the base interval (30 months) due to unsatisfactory LLRT performance."

The NRC has also provided the following concerning the extension of ILRT intervals to 15 years in NEI 94-01, Revision 3-A, NRC SER Section 4.0:

> "The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time."

3.1.2 Current CPNPP 10 CFR 50 Appendix J Requirements

Title 10 CFR Part 50, Appendix J was revised, effective October 26, 1995, to allow licenses to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance Based Requirements." On June 13, 1996 the NRC approved License Amendment No. 51 for CPNPP Unit 1 and Amendment 37 for Unit 2 authorizing the implementation of 10 CFR Part 50, Appendix J, Option B for Type A, B and C tests (Reference 11). Current Technical Specification 5.5.16 requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR Part 50 Appendix J, Option B, as modified by approved exemptions. The program is required to be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Guide 1.163 endorses, with certain exceptions, NEI 94-01 Revision 0 as an acceptable method for complying with the provisions of Appendix J, Option B.

Regulatory Guide 1.163, Section C.1 states that licensees intending to comply with 10 CFR Part 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 6) rather than using test intervals specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994. Nuclear Energy Institute 94-01, Section 11.0 refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of the Option B performance based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Types A, B, and C tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 7). The evaluation documented in NUREG-1493 included a study of the dependence or reactor accident risks on containment leak tightness for differing types of containment types, including a reinforced, shallow domed concrete containment similar to CPNPP containment structures. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three tests per ten years to one test per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

3.1.3 CPNPP 10 CFR 50 Appendix J Option B Licensing History

June 13, 1996
The NRC approved License Amendment No. 51 for CPNPP Unit 1 and Amendment 37 for Unit 2. The amendments revised the Technical Specifications to reflect the approval for the use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50, Appendix J, Option B for CPNPP Units 1 and 2. Implementation of the new performance based leakage rate testing program was based on the guidance provided by Regulatory Guide 1.163, September 1995. (ML02182034)(Reference 11)

September 5, 2000
The NRC approved License Amendment No. 79 for CPNPP Unit 1 and Amendment No. 79 for Unit 2. The amendments revised Technical Specifications to allow certain reactor containment building penetrations to be open during refueling activities under appropriate administrative controls. Specifically, this revision fully adopted the NRC-approved Technical Specification Traveler TSTF-312, Revision 1. (ML003747708)(Reference 12)

August 15, 2002
The NRC approved License Amendment No. 98 for CPNPP Unit 1 and Amendment No. 98 for Unit 2. The amendments revise TS 5.5.16, "Containment Leakage Rate Testing Program," extended 10 CFR Part 50, Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT) date for CPNPP, Units 1 and 2. The CPNPP Unit 1 date was extended from the fall of 2002 to December 2008, and Unit 2 was extended from the fall of 2006 to December 2012. (ML021970215)(Reference 13)

March 5, 2004
The NRC approved License Amendment No. 111 for CPNPP Unit 1 and Amendment 111 for Unit 2. The amendments revised TS requirements to permanently except seven containment isolation valves in each unit, in the residual heat removal and the containment spray systems, from the local leakage rate testing requirements of 10 CFR Part 50, Appendix J. (ML040690358)(Reference 14)

April 13, 2005
The NRC approved License Amendment No. 116 for CPNPP Unit 1 and Amendment 116 for Unit 2. The amendments revised TS requirements to extend the interval between local leakage rate tests for the containment purge and vent valves with resilient seats (containment purge valves, hydrogen purge valves, and containment pressure relief valves). The test intervals were extended from the current 184 days to 18 months between tests for all three types of valves and the "within 92 days after opening the valves" requirement was deleted. (ML050540419)(Reference 15)

December 13, 2007
The NRC approved License Amendment No. 141 for CPNPP Unit 1 and Amendment 141 for Unit 2. The amendments revise CPNPP, Units 1 and 2, TS requirements associated with the Containment Leakage Rate Testing Program (TS 5.5.16) to be consistent with 10 CFR Part 50, Section 55a, paragraph (g)(4). (ML073120252)(Reference 16)

3.1.4 Continued Acceptability of TS Amendment 141 for CPNPP Unit 1 and Unit 2

By the letter dated December 19, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070580126), TXU Generation Company LP (subsequently renamed Luminant Generation Company LLC, the licensee), submitted a license amendment request (LAR 06-010) to revise Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," for Comanche Peak Steam Electric Station (CPNPP), Units 1 and 2 (Reference 16). The requested amendment proposed to revise TS 5.5.16 for consistency with the requirements of paragraphs 50.55a(g)(4) of Title 10 of the Code of Federal Regulations (10 CFR), for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection (ISI) requirements in accordance with Division 1 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsections IWE and IWL. The proposed change was based on TS Task Force (TSTF)-343, Revision 1, which allowed the 10 CFR 50, Appendix J, Option B, visual examinations of the containment to be performed in accordance with ASME Code, Section XI, Subsections IWE and IWL, and meet the intent of visual examinations required by Regulatory Position C.3 of Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, without requiring additional visual examinations pursuant to RG 1.163.

Technical Specification Amendment 141 revised TS 5.5.16 by adding the following exceptions to RG 1.163, "Performance-Based Containment Leak-Test Program":

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements and frequency specified by ASME Code, Section XI, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements and frequency specified by ASME Code, Section XI, Subsection IWE, except where relief has been authorized by the NRC.

NRC SER Section 3.2, Evaluation, stated the following as the basis for acceptability of the requested TS Amendment:

> The CPNPP, Units 1 and 2, TS requirements for the Containment Leakage Rate Testing Program specify that the program shall be in accordance with the guidelines contained in RG 1.163, dated September 1995. Regulatory Position C.3 of this RG states: "Section 9.2.1, 'Pretest Inspection and Test Methodology,' of NEI [Nuclear Energy Institute] 94-01 provides guidance for visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination, except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.
>
> In addition to the requirements of RG 1.163 and NEI 94-01, per the requirements of 10 CFR 50.55a(g)(4), the concrete surfaces of the containment must be visually examined in accordance with ASME Code, Section XI, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with ASME Code, Section XI, Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every 5 years (in general, two times in a 10-year interval), and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year interval. The visual examination performed pursuant to Subsection IWL may be performed any time during power operation or during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this is the only time that the liner plate is fully accessible. The licensee substantiated that the requirements for visual examinations performed pursuant to Subsections IWE and IWL are more rigorous than those performed pursuant to RG 1.163 and NEI 94-01. The staff agrees that the combination of the Code requirements for rigor of the visual examinations plus the required third-party review will more than offset the fact that one fewer visual examination of the concrete surfaces will be performed during a 10-year interval. The fact that the exterior concrete visual examinations pursuant to Subsection IWL may be performed during power operation as opposed to during a refueling outage will have no effect on the quality of the examination; however, it provides flexibility in scheduling the visual examination.

Acceptability of TS Amendment 141

Section 9.2.3.2, Supplemental Inspection Requirements, of NEI 94-01, Revision 3-A, dated July 2012, and Revision 2-A, dated October 2008, both incorporated the inspection requirements of RG 1.163 Regulatory Position C.3 and expanded upon this requirement for ILRT intervals of up to fifteen years in Section 9.2.3.2, Supplemental Inspection Requirements, as follows:

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

ASME Section XI, Subsection IWE

ASME Section XI, Subsection IWE requires the performance of a complete visual examination of the containment liner once each inspection period. If the interval for the Type A test has been extended to 15 years (a minimum of 4 CISI Inspection Periods) a minimum of four complete inspections of the containment liner will be performed during the interval between ILRT performances. This frequency exceeds both the initial requirement of RG 1.163 and NEI 94-01 Section 9.2.3.2 as proposed by TS Amendment 141.

ASME Section XI, Subsection IWL

ASME Section XI, Subsection IWL requires the performance of a complete visual examination of the containment exterior concrete once every five years. If the interval for the Type A test has been extended to 15 years a minimum of two complete inspections of the containment exterior concrete will be performed during the interval between ILRT performances. This requirement when coupled with the following commitment provided in SER Section 3.2 will require the performance of three containment exterior concrete inspections if the interval for the Type A test has been extended to 15 years.

> Prior to performing an ILRT, the licensee will schedule its IWE and IWL examinations in a way that it be counted as a pre-ILRT examination.

SER Section 3.2 stated the following regarding the acceptability of the use of Subsection IWL:

> The staff agrees that the combination of the Code requirements for rigor of the visual examinations plus the required third-party review will more than offset the fact that one fewer visual examination of the concrete surfaces will be performed during a 10-year interval.

The result will be a requirement to perform a minimum of three visual examinations of the containment exterior concrete if the interval for the Type A test has been extended to 15 years. This satisfies the original intent of the SER in that one fewer visual examination of the concrete surfaces will be performed, with 3 concrete inspections during a 15-year interval instead of the original 2 concrete inspections in a 10-year interval.

Conclusion

It is the position of Luminant that it will continue to be acceptable to perform the visual inspections of the containment interior and exterior in accordance with the requirements and frequency specified by ASME Code, Section XI, Subsections IWE and IWL following the approval of the proposed TS Amendment. Luminant will also continue to meet the following commitment as provided by the NRC in Section 3.2 of the SER:

> Prior to performing an ILRT, the licensee will schedule its IWE and IWL examinations in a way that it be counted as a pre-ILRT examination.

No further evaluation of TS Amendment 141 in regards to the proposed activity is required.

3.2 Containment Building Description

The Reactor Containment structure is a fully continuous, steel-lined, reinforced concrete structure. It consists of a vertical cylinder and a hemispherical dome and is supported on an essentially flat foundation mat with a reactor cavity pit projection. The Containment superstructure is independent of the adjacent interior and exterior structures. Sufficient space is provided between the Containment and the adjacent structures to prevent contact under all combinations of loadings.

3.2.1 Dimensions of Containment:

- Inside diameter (ID): 135 ft. 0 in.
- Height of cylinder (top of foundation mat to dome spring line): 195 ft. 0 in.
- Inside radius of hemispherical dome: 67 ft. 6 in.
- Thickness of cylindrical walls: 4 ft. 6 in.
- Thickness of dome: 2 ft. 6 in.
- Foundation mat thickness: 12 ft. 0 in.
- Top of the foundation mat: approximately 4 ft. 6 in. below grade
- Containment design pressure: 50 psig
- Containment design temperature: 280 °F

3.2.2 Steel Liner

The entire inside face of the Containment (mat, walls, and dome) is lined with a continuous welded steel liner plate, attached with anchors to the reinforced concrete, to ensure a high degree of leaktightness. The thickness of the liner in the wall is 3/8 in. and in the dome is 1/2 in. A 1/4 in. thick plate is used on top of the foundation mat and is covered with a layer of concrete. Local thickened liner plate sections are provided at penetrations, major pipe and duct support attachments and at crane and rotating platform girder brackets, and at the bottom of the cylindrical wall's steel liner. Overlay plates and/or structural shapes may be used on the interior side of the liner for support of minor pipes and ducts, conduits, cable trays, and equipment.

Leak-chase channels are provided at liner seams, which, after construction, are inaccessible for other means of leaktightness examination.

3.2.3 Containment Penetrations and Attachments

A personnel airlock, an emergency airlock, and an equipment hatch provide access to the Containment structure. Containment airlocks are tested in accordance with 10CFR Part 50 Appendix J, Option B. A constant pressure of P_a is used to pressurize the volume between the airlock seals.

The personnel airlock is an approximately 9 ft. inside diameter electro-hydraulically operated double-door assembly. Each door is hinged and gasketed, with leakage test taps aligned to the annulus between the gasket sealing surfaces. Both doors are interlocked so that if one door is open, the other cannot be activated. Both doors are also furnished with hydraulic actuated as well as manual pressure equalizing valves, which can be operated by persons leaving or entering the personnel hatch. The personnel airlock has provisions for test pressurization at a pressure of Pa of the space between the two grooves at both ends of the airlock as well as provisions for pressurization at a pressure of Pa of the volume between the airlock doors. The doors are designed to maintain their functional capability during testing with no additional requirements for blocking beyond normal locking procedure.

The emergency airlock is an approximately 5-ft 9-in. inside diameter manually operated double-door assembly, with 2-ft 6-in. diameter doors. Both doors of the emergency airlock are furnished with manually operated pressure-equalizing connection and valves which are interlocked with the door operating mechanism and serve to equalize differential pressure across locked doors. The reactor building to airlock door (interior) requires installation of strongbacks for the performance of the overall leakage check. Other operating features are similar to those of the personnel airlock described previously.

The equipment hatch is a 16-ft 0-in. ID single closure penetration. The bolted hatch cover is mounted on the inside of the Containment, and is double gasketed with a leakage test tap between the gaskets. The hatch cover is provided with a hoist for handling.

Other smaller penetrations through the Containment include the main steam and feedwater lines, hot and cold pipes, the fuel transfer tube, and electrical conductors. All penetration sleeves are welded to the liner and anchored into the reinforced concrete Containment wall.

A fuel transfer tube penetration is provided for fuel transfer between the refueling canal in the Containment structure and the spent fuel pools in the Fuel Building. The penetration consists of a 20 in. stainless steel pipe inside a carbon steel sleeve. The inner pipe acts as the transfer tube; the outer tube is welded to the Containment liner. Bellows expansion joints are provided to permit differential movements. The fuel transfer tube is equipped with a bolted blind flange with double O-ring seals inside the containment. A test connection is provided so that the space between the transfer tube and the sleeve with connecting bellows can be pressurized to Pa in order to measure the leakage rate of the bellows or attachment welds.

Header plate type penetrations are used for electrical conductors passing through the Containment. The penetration header plate with double O-ring gaskets is bolted to a weld neck flange, which is welded to a steel penetration sleeve. The steel penetration sleeves are welded to the Containment vessel liner.

The Containment recirculation sump penetrations consist of sleeves embedded in the Containment mat with the process pipe seal welded to the sleeve by a seal ring inside the Containment. The sleeve is welded to the Containment liner. Each isolation valve outside the Containment is enclosed within a valve isolation tank, which is sealed to the sleeve by a 24-inch guard pipe and to the process pipe downstream of the isolation valve by a bellows expansion joint. The bellows, guard pipe and isolation tank assembly do not require type B testing.

All other mechanical penetrations do not incorporate any expansion joints or resilient seals. They consist of either a pipe embedded in the Containment wall concrete and welded to the Containment liner or a sleeve embedded and welded to the liner with the process pipe passing through the sleeve and sealed by a flued head welded to the sleeve. These penetrations are tested by a type C test performed on the isolation valves.

3.2.4 Containment Alternate Access for the Steam Generator and Reactor Pressure Vessel Head Replacement (Unit 1)

The Steam Generator (SG) and Reactor Vessel Head (RVH) Replacement Project created and restored a construction alternate access in the Containment Building (Containment Alternate Access) in accordance with administrative procedures and the design control program. The alternate access was used to facilitate the movement of original and replacement SGs and RVH out of and into the Containment Building. In accordance with the ASME Section XI repair/replacement program, the alternate access was restored consistent with the original containment specifications with any exceptions reconciled to the original specification.

Codes and Specifications

Restoration of the Containment Alternate Access was performed as a repair/replacement activity in accordance with the requirements of ASME Section XI, 1998 edition, 1999 and 2000 addenda.

The basic code for the restored Containment Building structure are appropriate portions of the Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973); ASME-ACI 359. The restored structure meets all applicable design loads and load combinations required by ASME-ACI 359.

Concrete placement, curing, and repair were in accordance with ACI 301-05. Concrete mix proportioning was per ACI 211.1-91 (reapproved 2002).

Project specification for restoration of the Containment Alternate Access address:

- Reinforcing steel procurement, testing and placement

- Cadweld® reinforcing steel splices procurement, testing and installation
- Concrete mix design, testing and placement
- Structural steel and materials procurement

Liner Restoration

The cut section of the Containment Building liner plate was rewelded to the liner plate with a full penetration weld. The new liner plate seam welds were examined using NDE methods specified within CC-5520. Liner weld was leak tested by vacuum box test method to satisfy leaktightness requirements of NRC Regulatory Guide 1.19.

Replacement material was purchased for Nelson studs in accordance with the requirements of the original plant specification for the Unit 1 liner plate.

Reinforcing Steel Restoration

The reinforcing steel bars cut during the creation of the Containment Alternate Access were reinstalled using Cadweld® splices or welding, as required. Reinforcing steel bars that were damaged during the creation of the access were repaired in accordance with AWS D1.4-98 or were replaced with reinforcing steel procured in accordance with the project specification. New No. 6 and No. 18 reinforcing steel used for the Containment Building wall restoration conform to ASTM A615 Grade 60 or ASTM A706 and meet or exceed the additional physical and chemical composition requirements described in UFSAR Section 3.8.1.6.2 for the Containment Building structure existing reinforcing steel.

Concrete Restoration

The concrete removed from within the Containment Alternate Access was restored with fresh concrete with a specified 28-day compressive strength of 4000 psi. Fresh concrete was qualified, tested, mixed, and placed in accordance with the project specification.

All repair activities were followed by the successful performance of the scheduled Type A test.

3.3 Integrated Leakage Rate Testing History (ILRT)

Previous Type A tests confirmed that the CPNPP reactor containment structures have leakage well under acceptance limits and represents minimal risk to increased leakage. Continued Type B and Type C testing for direct communication with containment atmosphere minimize this risk. Also, the Inservice Inspection (ISI) program and maintenance rule monitoring provide confidence in containment integrity.

Tables 3.3-1 and 3.3-2 list the past Type A ILRT results for CPNPP Unit 1 and 2 respectively.

Table 3.3-1 Unit 1 Type A ILRT History

Test Date	As-Found Leakage Rate (Containment air weight %/day)	As Left Leakage Rate (Containment air weight %/day)
07/04/1989 (Pre-Operation)	0.023	0.025
12/07/1993	0.05638	0.0557
04/14/2007	0.063019	0.0630

Table 3.3-2 Unit 2 Type A ILRT History

Test Date	As-Found Leakage Rate (Containment air weight %/day)	As Left Leakage Rate (Containment air weight %/day)
09/10/1992 (Pre-Operation)	0.052	0.047
12/01/1997	0.0317	0.0321
10/09/2012	0.0595	0.0594

The commercial operation dates for Unit 1 and Unit 2 are August 13, 1990, and August 3, 1993, respectively.

CPNPP TS 5.5.16.b, states "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig", and

CPNPP TS 5.5.16.c, states "The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day".

4.0 TECHNICAL ANALYSIS

4.1 Traditional Engineering Considerations

Consistent with the defense-in-depth philosophy discussed in Regulatory Guide 1.174, CPNPP has assessed other non-risk based considerations relevant to the proposed amendment. CPNPP has multiple inspections and testing programs that ensure the containment structure remains capable of meeting its design functions and that are designed to identify any degrading conditions that might affect that capability. These programs are discussed below.

4.1.1 Protective Coatings Program

The CPNPP Protective Coatings Program is implemented in accordance with engineering procedures. The procedure prescribes performing and documenting a complete visual inspection of coated surfaces (inclusive of the Liner Plate and Dome) within the Containment building. Recognition of degradation mechanisms as prescribed in EPRI Report 1003102, "Guideline on Nuclear Safety-Related Coatings", Revision 1 (formerly TR-109937) (Reference 17) are included in the Engineering Coatings Program. Frequency of inspection is conducted at a minimum once each fuel cycle. Items or areas, which cannot receive close visual examination, are examined from the best available vantage point using optical aids such as binoculars. Scaffolding and supplemental lighting are used, as required, in areas of particular interest. Items or areas, which are inaccessible to monitoring activities due to physical constraints or ALARA concerns, are documented. Inspections of containment coatings are performed each refueling outage.

The Unqualified Coatings Log (UCL) is utilized to track items previously identified as representing Design Basis Accident (DBA) unqualified coatings. The tracking process implementation, specifically, adding and deleting items from the log, is executed by the Protective Coatings Coordinator (PCC).

Potential UCL entries originate by the identification of circumstances where a coating is determined or suspected to be DBA unqualified by virtue of:

1. Having degraded in service, resulting in identification of a deficient condition.

2. Having been judged unqualified by virtue of an engineering assessment of an existing coating.

3. Having been applied in a manner not in compliance with the PCP.

Surfaces that are or are expected to become uncoated will also be tracked.

The UCL will be revised to reflect unqualified coating areas that have been reworked and inspected and are now acceptable. It will also reflect other unqualified coating areas that have been identified as requiring remediation in future outages. The UCL will be revised after every outage. The UCL revision process is tracked in the Corrective Action Program.

4.1.2 Containment Inservice Inspection Program

The Containment Inservice Inspection (CISJ) Program Plan details the requirements for the examination and testing of Class MC and Class CC components in accordance with ASME Section XI and 10CFR50.55a at CPNPP Units 1 and 2.

This CISI Program Plan covers the ten-year interval from September 10, 2012 to September 9, 2021 for Subsection IWE and IWL activities. This is the third interval for the Containment Inservice Inspection Program. Because the second interval of the CISI Program was extended to cover eleven (11) years, from September 10, 2001 to September 9, 2012 as allowed by IW A-2430, the third interval CISI Program has been shortened to include nine (9) years. The first interval CISI Program was conducted from September 9, 1996 to September 9, 2001

This Program Plan was developed in accordance with the 2007 Edition with the 2008 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by 10CFR50.55a. The requirement to implement the 2007 Edition with the 2008 Addenda of ASME Section XI is included in 10CFR50.55a(g)(4)(ii) and is based on an effective date of September 10, 2011, which was 12 months before the start of the third interval for the Containment Inservice Inspection Program.

The administrative procedures and inspection schedule described in the CISI program, combined with applicable CPNPP and approved vendor procedures, constitute the CISI portion of the ten-year ISI program. The third Interval CISI Program Plan dated September 2012 is currently in effect as of the date of this amendment request. The

Fourth Interval of the CISI Program Plan has not been developed at this time so all dates associated with the Fourth Interval are postulated. The Fourth Interval will be effective between September 10, 2021 and September 9, 2031.

Table 4.1.2-1, CPNPP Unit 1 IWE Examination Schedule

3rd Interval Period 1	9/10/2012 to 9/9/2015
	Planned Outages 1RF16/Spring 2013 1RF17/Fall 2014
3rd Interval Period 2	9/10/2015 to 9/9/2018
	Planned Outages 1RF18/ Spring 2016 1RF19/Fall 2017
3rd Interval Period 3	9/10/2018 to 9/9/2021
	Planned Outages 1RF20/ Spring 2019 1RF21/Fall 2020
4th Interval Period 1	9/10/2021 to 9/9/2024
	Planned Outages 1RF22/Spring 2022 1RF23/Fall 2023
4th Interval Period 2	9/10/2024 to 9/9/2028
	Planned Outages 1RF24/Spring 2025 1RF25/Fall 2026 1RF26/Spring 2028
4th Interval Period 3	9/10/2028 to 9/9/2031
	Planned Outages 1RF27/Fall 2029 1RF28/Spring 2031

Table 4.1.2-2, CPNPP Unit 2 IWE Examination Schedule

3rd Interval Period 1	9/10/2012 to 9/9/2015
	Planned Outages 2RF14/ Spring 2014
3rd Interval Period 2	9/10/2015 to 9/9/2018
	Planned Outages 2RF15/ Fall 2015 2RF16/ Spring 2017
3rd Interval Period 3	9/10/2018 to 9/9/2021
	Planned Outages 2RF17/Fall 2018 2RF18/ Spring 2020
4th Interval Period 1	9/10/2021 to 9/9/2024
	Planned Outages 2RF19/Fall 2021 2RF20/Spring 2023
4th Interval Period 2	9/10/2024 to 9/9/2028
	Planned Outages

	2RF21/Fall 2024 2RF22/Spring 2026 2RF23/Fall 2027
4th Interval Period 3	9/10/2028 to 9/9/2031
	Planned Outages 2RF24/Spring 2029 2RF25/Fall 2030

4.1.2.1 IWE/Accessible/Inaccessible Areas

Accessible areas are those areas of the metal containment that allow access for personnel and equipment to perform a direct visual examination or a remote examination utilizing magnifying aids. Drawings showing areas that have been previously defined by the CPNPP Coatings Program as "limited access areas" are contained in Appendices E and F of the CISI Program Plan. Areas considered as limited access by the coatings program also are considered to have limited access for visual examination personnel and equipment. These areas were further assessed and have been designated as inaccessible areas by the CISI Program Plan. An evaluation of the acceptability of inaccessible areas is required when conditions exist in accessible areas that could indicate degradation could also exist or could have extended into the inaccessible areas.

CPNPP shall provide the following in the Owners Activity Report (OAR-1), as required by 10 CFR 50.55a(b)(2)(ix)(A):

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation, and;
- A description of necessary corrective actions.

CPNPP has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, Luminant actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to CPNPP. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

4.1.2.2 Moisture Barriers and Liner Leak Chase Channel Test Piping

Moisture Barriers including seismic barrier material attached to containment liner and the plug-to-coupling pipe surface of the liner leak chase channel test piping is subject to a General Visual Inspection of 100% of the surfaces each CISI Inspection Period.

4.1.2.3 IWE/Augmented Examinations

Surface areas likely to experience accelerated degradation and aging require augmented examination per IWE-1240. Areas previously identified by the CPNPP Coatings Program as "areas/items of specific interest" are also considered to require special attention for

CISI examination. These areas will be further assessed during the third interval of examinations and may be designated as augmented areas by the CISI Program Plan. An evaluation of the acceptability of inaccessible areas is required when conditions exist in augmented areas that could indicate degradation could also exist or could have extended into the inaccessible areas.

There are currently no identified areas requiring Augmented Inspections at CPNPP.

4.1.2.4 Implementation of the Subsection IWL Program

Implementation of the Subsection IWL Program, from a schedule standpoint, is driven by:

- IWL-2400, INSPECTION SCHEDULE which requires concrete examinations to be performed at 1, 3, and 5 year frequency following the containment Structural Integrity Test and every 5 years thereafter per IWL-2410.
- IWL-2410(c) requires the 10-year and subsequent examinations of concrete to commence not more than 1 year prior to the specified dates and not more than 1 year after such dates.

Because CPNPP Units 1 and 2 are beyond ten years of commercial operations, a five-year frequency of concrete exams, plus or minus one year, is utilized.

The current schedule for the IWL examinations at CPNPP is summarized in Table 4.1.2-3 below:

Table 4.1.2-3, CPNPP IWL Examination Schedule

CPNPP Unit 1
1st 5 year Examination Concrete Exams: March 2014
2nd 5 year Examination Concrete Exams: March 2019
3rd 5 year Examination Concrete Exams: March 2024
4th 5 year Examination Concrete Exams: March 2029
5th 5 year Examination Concrete Exams: March 2034
CPNPP Unit 2
1st 5 year Examination Concrete Exams: March 2014
2nd 5 year Examination Concrete Exams: March 2019
3rd 5 year Examination Concrete Exams: March 2024
4th 5 year Examination Concrete Exams: March 2029
5th 5 year Examination Concrete Exams: March 2034

4.1.2.5 IWL/Accessible/Inaccessible Areas

Accessible areas are those areas of the concrete containment that allow access for personnel and equipment to perform a direct visual examination or a remote examination utilizing magnifying aids from existing floors, roofs, platforms, walkways, ladders, ground surface or other permanent vantage points. Concrete surface areas will be further assessed during the third interval examinations and may be designated as inaccessible areas by the CISI Plan. Inaccessible areas are exempt from examination per IWL-1220(b). An evaluation of the acceptability of inaccessible areas is required when conditions exist in accessible areas that could indicate degradation could also exist or could have extended into the inaccessible areas.

4.1.2.6 IWL/ Suspect Areas

Surface areas experiencing or likely to experience accelerated degradation and aging require special attention. These areas will be further assessed during third interval examinations and may be designated as suspect areas by the CISI Program Plan. An evaluation of the acceptability of inaccessible areas is required when conditions exist in suspect areas that could indicate degradation could also exist or could have extended into the inaccessible areas.

4.1.2.7 CISI Program Relief Requests

In cases where CPNPP has determined that ASME Section XI, Subsections IWE and IWL requirements are impractical to implement or has determined an alternative inspection approach to that specified in ASME Section XI would offer an acceptable (or equivalent) level of quality and safety, a 10CFR50.55a Request has been prepared and submitted to the NRC in accordance with 10CFR50.55a(g)(5), 10CFR50.55a(a)(3)(i) or 10CFR50.55a(a)(3)(ii), as applicable.

The following CPNPP Relief Request was submitted to the NRC for the Third CISI Interval.

> Relief Request 1/2E3-1 was submitted on October 31, 2013 with approval received April 10, 2014. The relief request addressed the "General Visual Examination of Electrical Penetrations with Radiant Energy Shielding."

4.1.2.8 Adopted Section XI Code Cases

In accordance with 10 CFR 50.55a(b)(5), ASME Section XI Code Cases that have been determined to be suitable for use in ISI Program Plans by the NRC are listed in referenced revisions of Regulatory Guide 1.147 "Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1". The use of other Code cases (than those listed in Regulatory Guide 1.147) may be authorized by the Director of the office of Nuclear Reactor Regulation upon request pursuant to 10 CFR 50.55a(a)(3). There have been no Code Cases adopted by CPNPP for the Third CISI Program Interval.

4.1.3 Plant Operational Performance

During power operation, instrument air leaks from air-operated valves inside containment and pressurizes the containment building. Containment pressure is monitored and conditions approaching the limits allowed by the Technical Specifications are annunciated. Because it is routinely necessary to reduce the increase in the building internal pressure by periodic operation of the containment pressure relief, a large pre-existing leak would make it unnecessary to periodically operate the containment pressure relief. Plant operators would notice this change in operating pattern.

Although not as significant as pressure resulting from a Design Basis Accident, the fact that the containment can be pressurized by leakage from air-operated valves provides a degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure). This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

4.1.4 Supplemental Inspection Requirements

Supplemental Inspections will not be required. Inspections of the exterior containment concrete surfaces and the steel liner plate inside containment will be conducted in accordance with TS 5.5.16.a (as modified by TS Amendment 141) (Reference 16) as follows:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

In addition to the above, the NRC in the SER for TS Amendment 141, which approved the above exceptions, incorporated the following commitment for CPNPP:

> Prior to performing an ILRT, the licensee will schedule its IWE and IWL examinations in a way that it be counted as a pre-ILRT examination.

The bases of acceptability for continued utilization of the examination requirements stated above is contained in Section 3.1.4.

4.1.5 Containment Leakage Rate Testing Program - Type B and Type C Testing Program

CPNPP Type B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B, and Regulatory Guide 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per Technical Specification 5.5.16, the allowable maximum pathway total Types B and C leakage is 0.6 L_a where 0.6 L_a equals approximately 151,000 sccm.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results from 2005 through 2014 for CPNPP Unit 1 and 2006 through 2014 for CPNPP Unit 2 has shown an exceptional amount of margin between the actual As-Found (AF) and As-left (AL) outage summations and the regulatory requirements as described below:

- The As-Found minimum pathway leak rate average for CPNPP Unit 1 shows an average of 5.74% of 0.6 L_a with a high of 7.36% or 0.044 L_a.
- The As-Left maximum pathway leak rate average for CPNPP Unit 1 shows an average of 15.07% of 0.6 L_a with a high of 19.62% or 0.118 L_a.
- The As-Found minimum pathway leak rate average for CPNPP Unit 2 shows an average of 5.22% of 0.6 L_a with a high of 7.17% or 0.043 L_a.
- The As-Left maximum pathway leak rate average for CPNPP Unit 2 shows an average of 12.40% of 0.6 L_a with a high of 15.85% or 0.095 L_a.

Tables 4.1.5-1 and 4.1.5-2 provide LLRT data trend summaries for CPNPP since 2005 for Unit 1 and 2006 for Unit 2 and encompasses both previous ILRTs. This summary shows that there has been no As-Found failure that resulted in exceeding the Technical Specification 5.5.16.d.1 limit of 0.6 L_a (151,000 sccm) and demonstrates a history of successful tests. The As-Found minimum pathway summations represent the high quality of maintenance of Type B and Type C tested components while the As-Left maximum pathway summations represent the effective management of the Containment Leakage Rate Testing Program by the program owner.

Table 4.1.5-1 Unit 1 Type B and C LLRT As-Found/As-Left Trend Summary

RFO	2005	2007	2008	2010	2011	2013	2014
AF Min Path (sccm)	7,974.50	8,532.35	10,877.85	11,116.55	7,422.55	7,497.20	6,530.80
Fraction of L_a	0.032	0.034	0.043	0.044	0.029	0.03	0.026
AL Max Path (sccm)	16,750.30	18,533.30	26,399.50	29,631.90	23,720.80	21,384.80	16,884.95
Fraction of L_a	0.066	0.074	0.105	0.118	0.094	0.085	0.067
AL Min Path (sccm)	8,042.50	8,417.05	10,551.65	11,116.55	7,422.55	8,107.20	6,759.13
Fraction of La	0.032	0.033	0.042	0.044	0.029	0.032	0.027

Table 4.1.5-2 Unit 2 Type B and C LLRT As-Found/As-Left Trend Summary

RFO	2006	2008	2009	2011	2012	2014
AF Min Path (sccm)	10,823.96	9,153.43	7,057.75	5,859.16	7,250.10	10,104.70
Fraction of L_a	0.043	0.036	0.028	0.023	0.029	0.04
AL Max Path (sccm)	23,823.41	23,941.51	16,509.92	13,720.81	19,674.20	19,846.20
Fraction of L_a	0.095	0.095	0.066	0.054	0.078	0.079
AL Min Path (sccm)	10,877.16	8,659.03	7,000.05	5,807.76	7,400.40	11,366.20
Fraction of L_a	0.043	0.034	0.029	0.023	0.029	0.045

The following Tables 4.1.5-3 and 4.1.5-4 identify the components that have not demonstrated acceptable performance during the previous two outages for CPNPP, Units 1 and 2 respectively:

Table 4.1.5-3, Unit 1 Type C LLRT Program Implementation Review

Component (1)	As-found SCCM	Admin Limit SCCM	As-left SCCM	Cause of Failure (2)	Corrective Action	Scheduled Interval
Pre-Spring 2013 (On Line Activities)						
1-8027	745	345	20	Seat Leakage	Valve Repaired (3)	30 month
Spring 2013						
1-8046	1060	1000	1060	Not Identified	Evaluated for continued service. (1)	30 month
Fall 2014						
1-HV-4166	585	518	585	Not Identified	Evaluated for continued service. (2)	30 month

(1) Work Order # 4612086 has been created to rework the valve to correct the leakage and ensure proper future operation. This work will be completed before or during 1RF17 (Fall 2014).

(2) Work Order # 4952592 has been created to rework the valve and ensure proper future operation. This work will be completed before or during 1RF18 (Spring 2016).

(3) Valve 1-8027 (Penetration 1-MV-0008) exceeded its administrative limit during 1RF15 (Fall 2011) and an evaluation determined it acceptable to leave in this leakage condition until rework prior to or during 1RF16 (Spring 2013) under work order # 4264698. Just prior to reworking the valve an As Found test was performed on the valve and it continued to fail to meet its administrative limit

value. Troubleshooting the leakage and repairs were performed under WO# 4266872 prior to 1RF16.

The percentage of the total number of Unit 1 Type B tested components that are on 120-month extended performance-based test intervals is 55%.

The percentage of the total number of Unit 1 Type C tested components that are on 60-month extended performance-based test intervals is 28%.

Table 4.1.5-4, Unit 2 Type C LLRT Program Implementation Review

Component (1)	As-found SCCM	Admin Limit SCCM	As-left SCCM	Cause of Failure (2)	Corrective Action	Scheduled Interval
Fall 2012						
None						
Spring 2014						
2-HV-5558	16156	345	20	Seat Leakage due to foreign material	Failed valve was replaced. (1)	30 month
2-HV-3486	2850	2070	2850	Not Identified	Evaluated for continued service. (3)	30 month
2-CA-0016	2450	2070	2450	Not Identified	Evaluated for continued service. (2)	30 month

(1) Work Order # 4715002 was initiated to rework/replace the valve before or during 2RF14. Work Order # 4770044 was initiated to perform a failure analysis on the valve.

Added a note / caution to the OWI-801 to inspect Leak Rate Monitor Test Rig fittings prior to installation to minimize the probability for the introduction of Foreign Materials.

(2) Work Order # 4804841 has been initiated to rework this valve at the next available opportunity.

(3) Work order # 4805001 has been initiated to rework 2-HV-3486 at the next available opportunity.

The percentage of the total number of Unit 2 Type B tested components that are on 120-month extended performance-based test intervals is 5%. (There are several that are on the 60-month extended interval, but only 1 on the 120-month extended interval).

The percentage of the total number of Unit 2 Type C tested components that are on 60-month extended performance-based test intervals is 29%.

4.1.6 NRC SER Limitations and Conditions

4.1.6.1 Limitations and Conditions Applicable to NEI 94-01 Revision 2-A

The NRC staff found that the use of NEI TR 94-01, Revision 2, was acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided the following conditions as listed in Table 4.1.6-1 were satisfied:

Table 4.1.6-1, NEI 94-01 Revision 2-A Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	CPNPP Response
For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1.)	CPNPP will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3.)	Reference Tables 4.1.2-1, 4.1.2-2 and 4.1.2-3 of this submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3.)	Reference Sections 4.1.2.1, 4.1.2.2 and 4.1.2.3 of this submittal.
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4.)	CPNPP Unit 1 steam generator replacements have been completed. Unit 2 replacements are not anticipated. There are no planned modifications for CPNPP Units 1 and 2 that will require a Type A test prior to the next Units 1 and 2 Type A test proposed under this LAR. There is no anticipated addition or removal of plant hardware within the containment building which could affect its leak-tightness.
The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2.)	CPNPP will follow the requirements of NEI 94-01 Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. In accordance with the requirements of 94-01 Revision 2-A, SER Section 3.1.1.2, CPNPP will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.

Limitation/Condition (From Section 4.0 of SE)	CPNPP Response
For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. CPNPP was not licensed under 10 CFR Part 52.

4.1.6.2 Limitations and Conditions Applicable to NEI 94-01 Revision 3-A

The NRC staff found that the guidance in NEI TR 94-01, Revision 3, was acceptable for referencing by licensees in the implementation for the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. However, the NRC staff identified two conditions on the use of NEI TR 94-01, Revision 3 (Reference NEI 94-01 Revision 3-A, NRC SER 4.0, Limitations and Conditions):

Topical Report Condition 1

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

Response to Condition 1

Condition 1 presents three (3) separate issues that are required to be addressed as follows:

- ISSUE 1 - The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.

- ISSUE 2 - In addition, a corrective action plan shall be developed to restore the margin to an acceptable level.

- ISSUE 3 - Use of the allowed 9-month extension for eligible Type C valves is only authorized for non-routine emergent conditions.

Response to Condition 1, Issue 1

The post-outage report shall include the margin between the Type B and Type C Minimum Pathway Leak Rate (MNPLR) summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 La.

Response to Condition 1, Issue 2

When the potential leakage understatement adjusted Type B and C MNPLR total is greater than the CPNPP administrative leakage summation limit of 0.50 La, but less than the regulatory limit of 0.6 La, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the CPNPP administrative limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

Response to Condition 1, Issue 3

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

Topical Report Condition 2

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

When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2

Condition 2 presents two (2) separate issues that are required to be addressed as follows:

- ISSUE 1 - Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.

- ISSUE 2 - When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2, Issue 1

The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25% in the LLRT periodicity. As such, CPNPP will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the As-Left leakage total for each Type C component currently on the 75-month extended test interval. This will result in a combined conservative Type C total for all 75-month LLRT's being "carried forward" and will be included whenever the total leakage summation is required to be updated (either while on line or following an outage). When the potential leakage understatement adjusted leak rate total for those Type C components being tested on a 75-month extended interval is summed with the non-adjusted total of those Type C components being tested at less than the 75-month interval and the total of the Type B tested components, if the MNPLR is greater than the CPNPP administrative leakage summation limit of 0.50 La, but less than the regulatory limit of 0.6 La, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the CPNPP administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

Response to Condition 2, Issue 2

If the potential leakage understatement adjusted leak rate MNPLR is less than the CPNPP administrative leakage summation limit of 0.50 La, then the acceptability of the 75-month LLRT extension for all affected Type C components has been adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

In addition to Condition 1, Parts 1, 2, which deal with the MNPLR Type B and C summation margin, NEI 94-01, Revision 3-A also has a margin related requirement as contained in Section 12.1, Report Requirements.

A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation *shall be identified in the report and a corrective action plan developed to restore the* margin to an acceptable level.

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

At CPNPP an adverse trend is defined as three (3) consecutive increases in the final pre-RCS Mode Change Type B and C MNPLR leakage summation values, as adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of La.

4.1.7 Nuclear Safety Advisory Letter (NASL) 14-2

Westinghouse transmitted Nuclear Safety Advisory Letter NSAL-14-2, Westinghouse LOCA Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties, to CPNPP via letter WPT-17781. The issue described in the NSAL was applicable to CPNPP as follows.

> This Nuclear Safety Advisory Letter (NSAL) is applicable to loss-of- coolant accident (LOCA) mass and energy (M&E) release calculations performed for Westinghouse-designed pressurized water reactors (PWRs) utilizing the methodology documented in WCAP-10325-P-A and WCAP-8264-P-A, Revision 1. The issue identified can potentially impact the plant specific LOCA M&E release calculation results, which are used as input to the containment integrity response analyses.
>
> The NSAL identifies that incorrect density and heat capacity values for the steam generator tubes were used in the analyses determining LOCA M&E release to the containment. Use of correct values in the current methodology results in an increase in the LOCA M&E releases, affecting the plant specific containment LOCA blowdown and post-blowdown transient conditions. The increase in the

LOCA M&E release has the potential to affect the following analyses. The impact on these analyses is also discussed.

The estimated containment pressure penalty for the issue listed in the NSAL is a maximum of 0.3 psi. The WCAP-10325-P-A and WCAP-8264- P-A, Revision 1 methodology contains modeling and initial condition assumption conservatisms that result in a calculated peak pressure that is 6 psi higher than the peak pressure that would be determined from a more realistic analysis. Approximately 2.5 psi of this 6 psi margin was used to address NSAL 11-5 as discussed in CPNPP Condition Report CR-2011-008432. This remaining 3.5 psi offsets the resulting increase in the containment peak pressure associated with the issue identified here. Therefore, if a more realistic analysis were performed peak containment pressure would not increase above the current value and there would be no impact on the 10CFR50, Appendix J, Type A, B, and C tests. Additionally, Technical Specification Bases (TSB) 3.6.1 documents the allowable containment leak rate based on a peak containment pressure (Pa) of 48.3 psig, which is greater than the FSAR Section 6.2 peak containment pressure of 47.0 psig. The 0.3 psi penalty associated with the issue identified in this CR when added to the FSAR Section 6.2 value does not challenge the TSB 3.6.1 value and does not challenge containment design pressure documented in TSB 3.6.4.

4.1.8 Containment Liner Degraded Coatings

The containment liner coatings for Units 1 and 2 have been identified as deficient from aging, possible application errors, and maintenance wear/damage.

In the early stages of performing the walkdowns for the Containment Coating Monitoring Report during 1RF14 in 2010, numerous areas of checking and cracking were being found on the liner plate with sizes never before seen. The walkdowns added 323.7 ft^2 of degraded coatings to the CCMR. In comparison to 2RF11, 22.40 ft^2 was added, and for 1RF13, 130 ft^2 was added. CR-2010-003462 was written to trend an abnormal increase in degraded coatings.

Emergency Sump Strainer OPERABILITY (required by TS 3.5.2, 3.5.3, and 3.6.6) is not affected. The strainers were qualified for all of the unqualified coatings in containment (ER-ESP-001, Rev. 2, Section 3.h).

During subsequent outages the Protective Coatings Program has been increasing margin (i.e., more square footage is being remediated than is being identified during the outage containment coatings inspections). CRs are written to track progress with regards to the recovery of the Containment Coatings and documenting the delta between the scheduled remediation and what was actually accomplished during each outage.

Degradation of the Containment Liner has not been identified as verified by CISI IWE examinations.

4.1.9 Information Notice (IN) 2010-12, "Containment Liner Corrosion"

This IN provides examples of containment liner degradation caused by corrosion. Concrete reactor containments are typically lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The reactor containment is required to be operable as specified in plant technical specifications to limit the leakage of fission product radioactivity from the containment to the environment. The regulations at 10 CFR 50.55a, "Codes and Standards," require the use of Subsection IWE of ASME Section XI to perform inservice inspections of containment components. The required inservice inspections include periodic visual examinations and limited volumetric examinations using ultrasonic thickness measurements. The containment components include the steel containment liner and integral attachments for the concrete containment, containment personnel airlock and equipment hatch, penetration sleeves, moisture barriers, and pressure-retaining bolting. The NRC also requires licensees to perform leak rate testing of the containment pressure-retaining components and isolation valves according to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as specified in plant technical specifications. This operating experience highlights the importance of good quality assurance, housekeeping and high quality construction practices during construction operations in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Corrosion to the containment liner is not a new industry issue. Programs and procedures are in-place to inspect the containment liner and would identify any areas subject to corrosion. Pre-job briefs heighten the awareness of the inspectors to industry issues on corrosion to the liners. The inspectors communicate effectively with each other.

To date, CPNPP has not identified any degradation of the containment liner, penetrations, hatches, or their pressure retaining bolting.

4.1.10 Information Notice (IN) 2014-07, "Degradation of Leak Chase Channel Systems For Floor Welds Of Metal Containment Shell And Concrete Containment Metallic Liner"

The U.S. Nuclear Regulatory Commission (NRC) issued this information notice (IN) to inform addressees of issues identified by the NRC staff concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures.

IN 2014-07 described the leak chase channel system as follows:

> Consists of steel channel sections that are fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection. Each test connection consists of a small carbon or stainless steel tube (less than 1-inch (2.5 centimeters) diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel.

IN 2014-07 describes a recessed box with a cover plate at floor level that allows for water to pool inside the recessed box and cause degradation. This configuration does not exist at CPNPP. Access to the compartmented air chambers is through ¼" diameter schedule 80 pipes with a threaded caps and these stand above the floor elevation. These pipes are routed within the arrangement of rebar and placed prior to concrete and are therefore integrated with the concrete floor.

CPNPP design is different than the design described in the IN. CPNPP performs visual inspections of the leak chase pipes to the CISI program which will occur 3 times in a 10 year period along with integrated leak rate tests to ensure leak tight integrity of the containment liner. No new actions are required to address this IN.

4.1.11 NRC Information Notice 92-20, Inadequate Local Leak Rate Testing

NRC Information Notice 92-20 was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

This is not applicable to CPNPP in that installed bellows assemblies, which are also Containment isolation barriers, are of the single ply design.

4.2 Evaluation of Risk Impact

4.2.1 Methodology

An evaluation has been performed to assess the risk impact for making the current "one-time" 15-year Type A Integrated Leak Rate Test (ILRT) interval permanent. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for Comanche Peak Unit 1 and Unit 2. The risk assessment follows the guidelines from NEI 94-01 (Reference 2), the methodology used in EPRI TR-104285 (Reference 8), the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 18), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 (Reference 5) as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion induced leakage of steel liners going undetected during the extended test interval (Reference 19), and the methodology used in EPRI 1009325, Revision 2-A (Reference 20).

In the SER issued by NRC letter dated June 25, 2008 (Reference 9), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. Table 4.2.1-1 addresses each of the four limitations and conditions for the use of EPRI 1009325, Revision 2.

Table 4.2.1-1 EPRI Report No.TR-1009325 Revision 2
Limitations and Conditions

Limitation/Condition (From Section 4.2 of SE)	Comanche Peak Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension	The technical adequacy of the CPNPP Unit 1 and Unit 2 PRA models are consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension. Reference 4.2.2 below.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and CCFP acceptance guidelines, and these guidelines have been used for the CPNPP plant specific assessments. The increase in population dose is 1.00E-02 person-rem/year for Unit 1 and 1.00E-02 person-rem/year for Unit 2. The increase in CCFP is 0.88% for Unit 1 and 0.90% for Unit 2. Both Unit 1 and Unit 2 prove to be below 1.5 percentage points and thus are considered to be very small.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 La as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the CPNPP plant specific risk assessment.
4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance	For CPNPP, containment over-pressure is NOT relied upon for emergency core cooling system (ECCS) performance.

4.2.2 Summary of PRA Quality for Permanent 15-Year ILRT Extension

Model Description

The CPNPP PRA model that was used for this application is a Level 1 and Level 2 analysis of Internal Events, including Internal Flood, for At-Power operation. An ASME PRA Standard compliant Fire PRA is in progress.

Model Background

The ASME PRA Standard (Reference 26), as endorsed by Regulatory Guide 1.200 Revision 2 (Reference 5), was used to demonstrate the technical adequacy of the CPNPP PRA model used for this application. The Peer Review of the CPNPP Revision 4 model is the baseline for PRA Standard compliance at CPNPP. PRA Standard compliance for PRA model updates and applications subsequent to Revision 4 is programmatically required at CPNPP.

The CPNPP PRA Model of Record (MOR) and documentation is in full compliance with the internal events portion of the ASME PRA Standard and Regulatory Guide 1.200. The MOR addresses Level 1 and Level 2 analysis of Internal Events, including Internal Flood, for At-Power operation. The model had a PWROG full scope Peer Review in March 2011. The Peer Review was performed against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (Reference 26) and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Regulatory Guide (RG) 1.200 Revision 2 (Reference 5). Further, the Peer Review was performed using the process defined in Nuclear Energy Institute (NEI) 05-04 (Reference 27). Immediately following the Peer Review, the model was revised to incorporate model and documentation changes in response to the Peer Review Findings & Observations (F&O) and issued as Revision 4A.

A minor periodic update was performed to bring the current Model of Record (MOR) for Comanche Peak Nuclear Power Plant to Revision 4B.

Identify Parts of the PRA that Conform to Capability Categories Lower Than Required for the Application

The current model of record, revision 4B meets Category II or better for all but 3 SR requirements. These remaining SR's were judged by the Peer Review Team to meet "Cat I". In Section 3.2.4.1 of the SER for EPRI TR-1009325, Revision 2 (Reference 9), the NRC staff states that Capability Category I of American Society of Mechanical Engineers (ASME) PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies. Therefore, the CPNPP PRA model is considered to be adequate to support the evaluation of changes to ILRT frequencies.

External Events Considerations

External hazards were evaluated in the CPNPP Individual Plant Examination for External Events (IPEEE) report, which was submitted in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4). The results of the CPNPP IPEEE are documented in the CPNPP IPEEE Main Report (Reference 28).

Luminant does not yet have quantifiable models for external hazards that meet the requirements of the ASME / ANS combined standard. A Fire PRA using the guidance in the ASME PRA Standard is in progress. A High Winds and Other Hazards PRA is planned to start following completion of the Fire PRA. Given that CPNPP's updated Ground Motion Response Spectrum (GMRS) is well below the SSE at all frequencies, seismic risk at the site is extremely unlikely to be a significant issue for any risk-informed application. The updated GMRS has allowed CPNPP to be one of the few plants that will submit a minimal Seismic risk evaluation in response to the 10CFR50.54f letter that was issued following the Fukushima Accident. Therefore, a Seismic PRA is not planned for CPNPP.

Summary

The CPNPP PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk informed applications. The complete evaluation of PRA Quality is provided in Attachment 7.

4.2.3 Summary of Plant-Specific Risk Assessment Results

The risk impact of permanently extending the Type A ILRT test frequency to once in fifteen years is as follows:

- The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 4.78E-08/yr for Unit 1 and 4.79E-08/yr for Unit 2. As such, the estimated change in LERF for Unit 1 and Unit 2 is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 (Reference 4) also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from External Events was also made. In this case, the total class 3b contribution to LERF including External Events was conservatively estimated as 2.83E-07/yr for Comanche Peak Unit 1 and 2.84E-07/yr for Comanche Peak Unit 2. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfies both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.00E-02 person-rem/ yr for Unit 1 and 1.00E-02 person-rem/yr for Unit 2. Note that this value is based on internal events only and does not consider external events. The total population dose is thus increased to 6.51 person-rem/yr for Unit 1 and 6.53 person-rem/yr for Unit 2. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose (6.51E-02 for Unit 1 and 6.53E-02 for Unit 2), whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 9). Moreover, the risk impact when compared to other severe accident risks is negligible. Note that CPNPP is below both criteria for meeting the definition of a very small population dose.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen-year interval is 0.88% for Unit 1 and 0.90% for Unit 2. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 9). Both Unit 1 and Unit 2 prove to be below 1.5 percentage points and thus are considered to be very small.

Therefore, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the Comanche Peak Unit 1 and Unit 2 risk profile.

4.2.4 Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B or Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for CPNPP confirm these general findings on a plant-specific basis considering the severe accidents evaluated for Comanche Peak, the Comanche Peak containment failure modes, and the local population surrounding Comanche Peak.

Details of the CPNPP Unit 1 and 2, risk assessment are contained in Attachment 6 of this submittal.

4.3 Conclusion

NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR Part 50, Appendix J, Option B. It incorporated the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. CPNPP is adopting the guidance of NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, for the CPNPP, Units 1 and 2, 10 CFR Part 50, Appendix J testing program plan.

Based on the previous ILRT tests conducted at CPNPP, Units 1 and 2, it may be concluded that the permanent extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR Part 50, Appendix J and the overlapping inspection activities performed as part of the following CPNPP, Units 1 and 2 inspection programs:

- Containment Inservice Inspection Program (IWE/IWL)
- Containment Coatings Inspection and Assessment Program

This experience is supplemented by risk analysis studies, including the CPNPP, Units 1 and 2 risk analysis provided in Attachment 6. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 15 years results in a very small change to the CPNPP, Units 1 and 2 risk profiles.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Luminant Generation Company LLC (Luminant) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment to the TS involves the extension of the CPNPP, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. The containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for CPNPP, of 1.00E-02 person rem/yr to 6.51 person-rem/yr for Unit 1 and 6.53 person-rem/yr for Unit 2 using the EPRI guidance with the base case corrosion included. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

As documented in NUREG-1493, Type B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The CPNPP, Units 1 and 2 Type A test history supports this conclusion.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and; (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that have already taken place so their deletion is solely an administrative action that has no effect on any component and no impact on how the units are operated.

Therefore, the proposed change does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment to the TS involves the extension of the CPNPP, Unit 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that would have already taken place by the time this amendment is approved; therefore, their deletion is solely an administrative action that does not result in any change in how the units are operated.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment to TS 5.5.16 involves the extension of the CPNPP, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

The proposed change involves only the extension of the interval between Type A containment leak rate tests and Type C tests for CPNPP, Units 1 and 2. The proposed surveillance interval extension is bounded by the 15-year ILRT Interval and the 75-month Type C test interval currently authorized within NEI 94-01, Revision 3-A. Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, TS and the Maintenance Rule serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A and Type C test intervals.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that would have already taken place by the time this amendment is approved; therefore, their deletion is solely an administrative action and does not change how the units are operated and maintained.

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

Based on the above, Luminant concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.2 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR Part 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type C test frequency will not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2, provided a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94 01, Revision 3-A, Section 9.2.3.1 states that Type A ILRT intervals of up to 15 years are allowed by this guideline. The Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI Report 1018243 (Formerly TR-1009325, Revision 2) indicates that, in general, the risk impact associated with ILRT interval extensions for intervals up to 15 years is small. However, plant-specific confirmatory analyses are required.

The NRC staff reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01, Revision 2, the NRC staff determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. This guidance includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC staff finds that the Type A testing methodology as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by NEI TR 94-01, Revision 2, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight.

In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

For EPRI Report No. 1009325, Revision 2, a risk-informed methodology using plant-specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC staff finds that the proposed methodology satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174. The NRC staff, therefore, found that this guidance was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.2 of the Safety Evaluation Report (SER).

The NRC staff reviewed NEI TR 94-01, Revision 3, and determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J, as modified by the conditions and limitations summarized in Section 4.0 of the associated Safety Evaluation. This guidance included provisions for extending Type C LLRT intervals up to 75 months. Type C testing ensures that individual containment isolation valves are essentially leak tight. In addition, aggregate Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. The NRC staff, therefore, found that this guidance, as modified to include two limitations and conditions, was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing. Any applicant may reference NEI TR 94-01, Revision 3, as modified by the associated SER and approved by the NRC, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, in a licensing action to satisfy the requirements of Option B to 10 CFR Part 50, Appendix J.

5.3 Precedent

This request is similar in nature to the following license amendments to extend the Type A Test Frequency to 15 years, as previously authorized by the NRC:

- Nine Mile Point Nuclear Station, Unit 2 (Reference 21)
- Arkansas Nuclear One, Unit 2 (Reference 22)
- Palisades Nuclear Plant (Reference 23)
- Virgil C. Summer Nuclear Station, Unit 1 (Reference 24)

This request is also similar in nature to the following license amendments to extend the Type A Test Frequency to 15 years, and the Type C Test Frequency to 75 months as previously authorized by the NRC:

- Surry Power Station, Units 1 and 2 (Reference 25)

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, September 1995.

2. NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 2012.

3. NEI 94-01, Revision 2-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, October 2008.

4. Regulatory Guide 1.174, Revision 2, An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis, May 2011.

5. Regulatory Guide 1.200, Revision 2, An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities, March 2009.

6. NEI 94-01, Revision 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 1995.

7. NUREG-1493, Performance-Based Containment Leak-Test Program, January 1995.

8. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

9. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), dated June 25, 2008, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (TAC No. MC9663).

10. Letter from S. Bahadur (NRC) to B. Bradley (NEI), dated June 8, 2012, Final Safety Evaluation of Nuclear Energy Institute (NEI) Report 94-01, Revision 3, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J (TAC No. ME2164).

11. Letter from T. Polich (NRC) to C. L. Terry (TU Electric), dated June 13, 1996. Comanche Peak Steam Electric Station, Units 1 and 2 - Amendment Nos. 51 and 37 to Facility Operating License Nos. NPF-87 and NPF-89 (TAC NOS. M94992 and M94993).

12. Letter from D. Jaffe (NRC) to C. L. Terry (TXU Electric), dated September 5, 2000. Comanche Peak Steam Electric Station, Units 1 and 2-Issuance of Amendments Re: Administrative Controls for Open Penetrations During Refueling Operations (TAC Nos. MA9071 and MA9072).

13. Letter from D. Jaffe (NRC) to C. L. Terry (TXU Energy), dated August 15, 2002. Comanche Peak Steam Electric Station, Units 1 and 2 - Issuance of Amendments Re: One Time Extension of Appendix J, Type A, Integrated Leak Rate Test Interval from Ten to Fifteen Years (TAC Nos. MB3685 and MB3686).

14. Letter from M. Thadani (NRC) to M. R. Blevins (TXU Energy), dated March 5, 2004. Comanche Peak Steam Electric Station, Units 1 and 2 - Issuance of Amendments Re: Revision to Technical Specification for Containment Isolation Valves (TAC Nos. MB8185 and MB8186).

15. Letter from M. Thadani (NRC) to M. R. Blevins (TXU Energy), dated April 13, 2005. Comanche Peak Steam Electric Station, Units 1 and 2 – Issuance of Amendments to Extend Surveillance Frequency for Containment Purge, Hydrogen Purge, and Containment Pressure Relief Valves (TAC Nos. MC0911 and MC0912).

16. Letter from B. Singal (NRC) to M. R. Blevins (TXU Energy), dated December 13, 2007. Comanche Peak Steam Electric Station, Units 1 and 2 – Issuance of Amendments Re: Revision to Technical Specification 5.5.16, Containment Leak Rate Testing Program (TAC Nos. MD4074 and MD4075).

17. EPRI Report 1003102, "Guideline on Nuclear Safety-Related Coatings", Revision 1 (formerly TR-109937).

18. Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

19. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.

20. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA: 2008.

21. Letter from R. V. Guzman (NRC) to S. L. Belcher (NMP), dated March 30, 2010, Nine Mile Point Nuclear Station, Unit No. 2 - Issuance of Amendment RE: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650).

22. Letter from N. K. Kalyanam (NRC) to Vice President, Operations (ANO), dated April 7, 2011, Arkansas Nuclear One, Unit No. 2 - Issuance of Amendment RE: Technical Specification Change to Extend Type A Test Frequency to 15 Years (TAC No. ME4090).

23. Letter from M. L. Chawala (NRC) to Vice President, Operations (PNP), dated April 23, 2012, Palisades Nuclear Plant - Issuance of Amendment to Extend the

Containment Type A Leak Rate Test Frequency to 15 Years (TAC No. ME5997).

24. Letter from S. Williams (NRC) to T. D. Gatlin (VCSNS), dated February 5, 2014, Virgil C. Summer Nuclear Station, Unit 1 - Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385).

25. Letter from S. Williams (NRC) to D. A. Heacock (VEPCO), dated July 3, 2014, Surry Power Station, Units 1 And 2-Issuance Of Amendment Regarding The Containment Type A And Type C Leak Rate Tests (TAC Nos. MF2612 and MF2613).

26. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, February 2009.

27. NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Nuclear Energy Institute, November 2008.

28. CPNPP IPEEE Report: ER-EA-008, "Individual Plant Examination of External Events for severe Accident Vulnerabilities - CPSES" dated June 1995. [Docketed as TU Electric Letter logged TXX-95171 from C.L. Terry to USNRC dated June 27, 1995]

ATTACHMENT 2 TO TXX-15001
PROPOSED TECHNICAL SPECIFICATION CHANGE (MARK-UP)
5.5-14

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. 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:

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

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

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

c. 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. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995"~~ as modified by the following exceptions:

NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008,

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

~~3. NEI 94-01 – 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)."~~

ATTACHMENT 3 TO TXX-15001
PROPOSED TECHNICAL SPECIFICATION BASES CHANGE (MARK-UP)
B 3.6-25

BASES

REFERENCES (continued)

8. BTP CSB 6-4.

9. DBD-ME-013.

10. 10 CFR 50, Appendix J, Option B.

~~11. Regulatory Guide 1.163 (September 1995).~~

ATTACHMENT 4 TO TXX-15001
RETYPED TECHNICAL SPECIFICATION PAGE
5.5-14

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. 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:

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

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

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

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

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

ATTACHMENT 5 TO TXX-15001
RETYPED TECHNICAL SPECIFICATION BASES PAGE
B 3.6-25

BASES

REFERENCES (continued)

8. BTP CSB 6-4.

9. DBD-ME-013.

10. 10 CFR 50, Appendix J, Option B.

ATTACHMENT 6 TO TXX-15001
CPNPP PRA EVALUATION
PERMANENT ILRT EXTENSION RISK ASSESSMENT

Comanche Peak Nuclear Power Plant

PRA Evaluation

Permanent ILRT Extension Risk impact Assessment

ECE 2.15 Evaluation Log # 242, Rev. 1

Risk Assessment and Applications

December 15, 2014

Preparer: ____________ Date: 12-16-2014

Derek Seaman

Reviewer: ____________ Date: 12-16-14

Dan Tirsun

Approval: ____________ Date: 12/16/2014

Betsy Luengas

PRA DOCUMENT CONTROL FILES	**LAN FOLDER** - CPRXEngr\R&R Evaluations and Reviews\Evaluation #242	**Pg** 2

PRA DOCUMENT NO.: Evaluation Log # 242 REV.: 1

DOCUMENT TITLE: Permanent ILRT Extension Risk impact Assessment

BASE MODEL— _ YES, or other (SPECIFY): Base PRA Input provided from MOR R4B, ECE 2.15 Evaluation # 241, Rev 0

REVISON.: 0

BASE VERIFICATION DETAILS (CDF / # CUTSETS)N/A_ OTHER:____

REVIEWED BY: Daniel Tirsun DATE: 12/16/14

ELECTRONIC FILE NAME ___N/A___________ __ YES, or other:_

FILE NAMES / DESCRIPTIONS (List all):

Unit 1 CPNPP ILRT Calculations (Base Case).xlsx

Unit 1 CPNPP ILRT Calculations(External Events Sensitivity).xlsx

Unit 1 CPNPP ILRT Calculations(Corrosion Sensitivity and Initial Failure Sensitivity).xlsx

Unit 1 CPNPP ILRT Calculations(Corrosion Sensitivity).xlsx

Unit 2 CPNPP ILRT Calculations (Base Case).xlsx

Unit 2 CPNPP ILRT Calculations (External Events Sensitivity).xlsx

Unit 2 CPNPP ILRT Calculations (Corrosion Sensitivity).xlsx

FILE LOCATION SAME AS BASE: _ YES X or other (SPECIFY): CPRXEngr\R&R Evaluations and Reviews\Evaluation #242\

TRUNCATION (same as ORAM Base model)?

_N/A__ YES: If YES, **skip to next section** __CDF

___ NO: If NO, identify limits here __LERF

PRA QUANT INPUT FILE: (SPECIFY): N/A

PRA QUANT CONTROL FILE: __N/A__QuantifierSettings.INI or other (SPECIFY): __

ADVANCED OPTIONS: _N/A __Quantification Job Aid or other (SPECIFY):__

FLAG SETTINGS: __N/A MASTERR4B.txt _YES __NO

or other (SPECIFY): ____________________ _ YES __NO

MUTUALLY EXCLUSIVE (MUTEX):_ or changes (SPECIFY): N/A

RECOVERY RULE FILE _ or other (SPECIFY): N/A

CUTSET FILES (list file names):N/A

FILE LOCATION SAME AS BASE: X YES NO (SPECIFY): ______

SOFTWARE LISTING _ other (SPECIFY): N/A

CODE: N/A VERSION: N/A DATE / EVAL# 9/17/14

FILE VERIFICATION COMPLETE:Daniel Tirsun__ DATE 12/16/14

Revision Record

Revision 1 to this document was issued to incorporate minor changes to the information provided by the responsible organization pertaining to coatings.

Revision 0 of this document was the original issue.

Contents

1 Purpose of Analysis

1.1 Purpose

The purpose of this analysis is to provide a risk assessment for making the current "one-time" 15 year Type A Integrated Leak Rate Test (ILRT) interval permanent. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for Comanche Peak Unit 1 and Unit 2. The risk assessment follows the guidelines from NEI 94-01 (Reference 1), the methodology used in EPRI TR-104285 (Reference 2), the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 3), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion induced leakage of steel liners going undetected during the extended test interval (Reference 5), and the methodology used in EPRI 1009325, Revision 2-A (Reference 20).

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A 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 rate was less than the limiting containment leakage rate of 1La[1].

The basis for the current fifteen year test interval is provided in Section 11.0 of NEI 94-01, Revision 3-A, and was established in 2008. 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.

The NRC report on performance-based leak testing, NUREG-1493, determined that for a representative PWR plant, containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is required to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for Comanche Peak Unit 1 and Unit 2.

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[1] La (percent/24 hours) is the maximum allowable leakage rate at pressure Pa (calculated peak containment internal pressure related to the design basis accident) as specified in the technical specifications.

The Guidance provided in Appendix H of EPRI Report No. 1009325, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,"* (Reference 20) for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

The Comanche Peak containment vessel is a fully continuous, steel-lined, reinforced concrete structure with a net free volume of approximately 2.985 x 10^6 ft^3 (Reference 26). The Comanche Peak Containment vessels are examined in accordance with the requirements of ASME Code Section XI, Subsection IWE, the plant protective coatings program, and Technical specifications.

> *The Containment Inservice Inspection (CISI) Program Plan details the requirements for the examination and testing of Class MC and Class CC components in accordance with ASME Section XI and 10CFR50.55a at the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2.*
>
> *This CISI Program Plan covers the ten-year interval from September 10, 2012 to September 9, 2021 for Subsection IWE and IWL activities. This is the third interval for the Containment Inservice Inspection Program. Because the second interval of the CISI Program was extended to cover eleven (11) years, from September 10, 2001 to September 9, 2012 as allowed by IWA-2430, the third interval CISI Program has been shortened to include nine (9) years. The first interval CISI Program was conducted from September 9, 1996 to September 9, 2001*
>
> *This Program Plan was developed in accordance with the 2007 Edition with the 2008 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by 10CFR50.55a. The requirement to implement the 2007 Edition with the 2008 Addenda of ASME Section XI is included in 10CFR50.55a(g)(4)(ii) and is based on an effective date of September 10, 2011, which is 12 months before the start of the third interval for the Containment Inservice Inspection Program.*
>
> *Section 2.0 of the Plan provides pertinent Section XI, regulatory and plant background information; Section 3.0 details the implementation of the Metal Containment CISI Program Plan, and Section 4.0 details the implementation of the Concrete Containment CISI Program Plan and Section 5.0 includes CISI Program Plan References.*
>
> *Examination of the Containment liner and penetrations (ASME Section XI, Section IWE) and exterior surface of the concrete Containment (ASME section XI, section IWL) are performed in accordance with procedures TX-ISI-IWE and TX-ISI-IWL, respectively.*

During activities that require repair of the containment liner, ASME Section XI, Subsection IWE requires visual exams to assess the condition of the containment liner metal surface for coatings exhibiting evidence of flaking, blistering, peeling, discoloration (these are usually defects characterized of a painted surface not a metal surface) and other signs of distress. Liner metal surfaces that are non-coated and require repair show signs of cracking, discoloration, and structural distortion (wear, pitting, corrosion, gouges, dents or other surface discontinuities.) Prior to any repair, an inspection is performed by NDE personnel to assess the condition of the base material. Following completion of repairs a final inspection is performed by NDE personnel to determine acceptability of the final condition and to act as a reference for future inspections (there is no requirement of an NDE inspection following coating repairs).

NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. 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.

1.3 Criteria

The acceptance guidelines in RG 1.174 (Reference 25) are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. 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. As the Comanche Peak Unit 1 and Unit 2 Level I PRAs do not credit containment features, the Type A test does not impact CDF. Therefore, the relevant risk metric is the change in LERF. RG 1.174 defines small changes in LERF as below 10^{-6} per reactor year. The criteria described below are taken from the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 25).

Regarding Conditional Containment Failure Probability (CCFP), the NRC concluded that a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one time fifteen year ILRT extension requests. To this end the NRC has endorsed a small increase in CCFP as an increase in CCFP be less than or equal to 1.5% (Reference 25).

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. The NRC concluded that for purposes of assessing the risk impacts of the Type A ILRT extension in accordance with the EPRI methodology, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive (Reference 25).

2 References

1. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, Revision 3-A, July 2012.
2. Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 2, May 2011.
5. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.
7. Evaluation of Severe Accident Risks: Surry Unit 1, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, December 1990.
8. Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
11. Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.
13. Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.
14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.
15. Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, December 1990.
16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
17. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.
18. Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.
19. Letter from D.E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
20. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA: 2008.

21. Letter from P.P. Sena III (FENOC) to Document Control Desk (NRC), dated June 18, 2009, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, LER 2009-003-00, "Containment Liner Through Wall Defect Due to Corrosion."

22. Letter from E.A. Larson (FENOC) to Document Control Desk (NRC), dated February 14, 2014, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, LER 2013-002-01, "Containment Liner Through Wall Defect Discovered During Planned Visual Inspection."

23. Letter from J.E. Pollock (AEP Indiana Michigan Power) to Document Control Desk (NRC), dated March 16, 2001, submitting LER 316/2000-001-01, "Through-Liner Hole Discovered in Containment Liner."

24. ER-EA-008 Individual Plant Examinations of External Events for Severe Accidents Vulnerabilities Comanche Peak Steam Electric Station, June 1995.

25. Final Safety Evaluation For NEI Topical Report 94-01 Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and EPRI Report No. 1009325 Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Test Intervals".

26. Comanche Peak Nuclear Power Plant Final Safety Analysis Report (FSAR) through Amendment 105.

27. ECE 2.15 Evaluation Log #212, Rev. 0, "IPEEE Seismic Analysis Applicability to Rev. 4A Model Results," July 2012.

28. R&R-PN-022, Rev. 4C, "Comanche Peak Nuclear Power Plant Accident Sequence Quantification, " October 17, 2013.

29. ECE 2.15, Evaluation Log #241, Rev. 0, "Comanche Peak Inputs Used for Integrated Leak Rate Test Interval Extension," June 4, 2014.

30. Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385), February 5th, 2014. Adams ML13326A204.

31. ECE 2.15 Evaluation Log #211. Rev. 0, "IPEEE High Wind Analysis Applicability to Rev. 4A Model Results," June 2012.

32. Comanche Peak Nuclear Power Plant, Docket Nos. 50-445 AND 50-446, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident.

33. Individual Plant Examination Submittal: Comanche Peak Steam Electric Station Volume II: Back-End Analysis, H.C. da Silva, Jr. October, 1992.

34. R&R-PN-022, Rev. 0, "Comanche Peak Steam Electric Station Probabilistic Safety Assessment – Evaluation of Risk Significance of ILRT Extension," November, 2001.

35. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Regulatory Guide 1.200, Revision 2, March 2009.

3 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2-A, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals"* (Reference 20), EPRI TR-104285 (Reference 2), NUREG-1493 (Reference 6) and the Calvert Cliffs liner corrosion analysis (Reference 5). The analysis uses results from the current Comanche Peak Unit 1 and Unit 2 Level 2 PRA models to establish frequency of fission product releases. Fission product release magnitudes are extrapolated from results of NUREG/CR-4551 (Reference 7) to account for plant specific characteristics. This risk assessment is applicable to Comanche Peak Unit 1 and Unit 2.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in Reference 20.

2. Develop plant specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 (Reference 4) and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact of the ILRT interval extension on the Conditional Containment Failure Probability (CCFP) and the population dose and compare with the acceptance guidance of Reference 20.

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore:

- Consistent with the other industry containment leak risk assessments, the Comanche Peak Unit 1 and Unit 2 assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- The evaluation for Comanche Peak Unit 1 and Unit 2 uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2-A, *"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals."*

4 Ground Rules

The following ground rules are used in the analysis:

- The technical adequacy of the Comanche Peak Unit 1 and Unit 2 PRA models are consistent with the requirements of Regulatory Guide 1.200 (Reference 28) as is relevant to this ILRT interval extension.

- The current Comanche Peak Unit 1 and Unit 2 Level 1 and Level 2 internal events PRA models are explicitly used in this analysis to assess fission product release frequencies.

- It is appropriate to use the Comanche Peak Unit 1 and Unit 2 internal events PRA models as gauges to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations; this is evaluated in the sensitivity analysis which uses available information from the Comanche Peak IPEEE (Reference 24).

- Dose results for the containment failures modeled in the PRA can be characterized by scaling information provided in NUREG/CR-4551 (Reference 7). Specifically, Comanche Peak population dose estimates are obtained by scaling the NUREG/CR-4551 reference plant results by differences in population, reactor power level (assumed proportional to fission product inventory), and nominal containment maximum leakage rate (La). Results of sensitivity studies are included which utilize Comanche Peak Unit 1 and 2 release class doses used in the one-time 15 year ILRT extension (Reference 34), modified to account for differences in ILRT methodology and appropriately adjusted for power level and population growth (see discussion in Section 6.4)

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology (Reference 2) and are summarized in Section 4.2.

- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.

- The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 (Reference 8 and Reference 9).

- The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (Reference 20).

- The Class 3b is very conservatively categorized as LERF based on the previously approved methodology (References 8 and 9).

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

5 Inputs

This section summarizes the general resources available as input (Section 5.1) and the plant specific resources required (Section 5.2).

5.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 (Reference 10)
2. NUREG/CR-4220 (Reference 11)
3. NUREG-1273 (Reference 12)
4. NUREG/CR-4330 (Reference 13)
5. EPRI TR-105189 (Reference 14)
6. NUREG-1493 (Reference 6)
7. EPRI TR-104285 (Reference 2)
8. NUREG-1150 (Reference 15) and NUREG/CR-4551 (Reference 7)
9. NEI Interim Guidance for One-Time Extension of ILRT (Reference 3, Reference 17)
10. Calvert Cliffs Liner Corrosion Analysis (Reference 5)
11. EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20)

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth studies provide ex-plant consequences analysis for a 50 mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for Comanche Peak Unit 1 and Unit 2. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent fifteen year extension of the ILRT interval.

5.1.1 NUREG/CR-3539 (Reference 10)

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 (Reference 16) as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

5.1.2 NUREG/CR-4220 (Reference 11)

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

5.1.3 NUREG-1273 (Reference 12)

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

5.1.4 NUREG/CR-4330 (Reference 13)

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

"...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

5.1.5 EPRI TR-105189 (Reference 14)

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

5.1.6 NUREG-1493 (Reference 6)

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

Reduction in ILRT frequency from three per ten years to one per twenty years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

5.1.7 EPRI TR-104285 (Reference 2)

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded: "... the proposed CLRT (containment leak rate tests) frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year..."

5.1.8 NUREG-1150 (Reference 15) and NUREG/CR 4551 (Reference 7)

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50 mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Comanche Peak Unit 1 and Unit 2 Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Comanche Peak. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

5.1.9 NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals (Reference 17)

The guidance provided in this document builds on the EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

5.1.10 Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension (Reference 5)

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. Licensees may consider approved LARs for one-time extensions involving containment types similar to their facility. Note that Calvert Cliffs is constructed with a steel lined concrete containment similar to Comanche Peak.

5.1.11 EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (Reference 20)

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant specific supplemental risk impact assessments and builds on the previous

EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the Comanche Peak Unit 1 and Unit 2 assessments to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Classes 3a and 3b scenarios in this analysis as described in Section 6. The Comanche Peak Unit 1 and 2 fifteen (15) year ILRT extension used an early version of this methodology.

5.2 Plant Specific Inputs

The plant specific information (Reference 29) used to perform the Comanche Peak Unit 1 and Unit 2 ILRT Extension Risk Assessments include the following:

- Level 1 Model results
- Level 2 Model results
- Release category definitions used in the Level 2 Model
- Population within a 50 mile radius for the year 2056 based on the Combined Operating License application FSAR for CPNPP Units 3 & 4. This represents the most recent projected growth for the area and by assuming positive population growth past the Unit 1 and Unit 2 (2030 and 2033 respectively) end of license this number conservatively bounds population estimates.
- Containment Fragility Curves

5.2.1 Level 1 Model

The Level 1 PRA models that are used for Comanche Peak Unit 1 and Unit 2 are characteristic of the as-built plants. The current Unit 1 Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 4.08E-06/yr using a truncation value of 1.00E-13. The current Unit 2 Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 4.08E-06/yr a truncation value of 1.00E-13. Both models account for an increased CDF due to internal flood.

5.2.2 Level 2 Model

The Level 2 Models that are used for Comanche Peak Unit 1 and Unit 2 were developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 5-1 and Table 5-2 summarize the pertinent Comanche Peak Unit 1 and Unit 2 results in terms of release category. Note that the enumerated total internal events Level 2 release frequency is slightly larger than that of the internal events CDF. This difference arises as a result of the numerical truncation issues resulting from the full integration of core damage end-states into the Level 2 model and the impact of the CAFTA small number approximation as applied to the detailed containment failure model. The small number approximation is a standard modeling practice. While this difference is observable, it does not significantly impact the results of the simplified Level 2 PRA or the associated conclusions drawn with regard to the ILRT extension.

Table 5-1: Comanche Peak Unit 1 and Unit 2 Level 2 LERF Release Categories and Frequencies

Release Category	Definition	Unit 1 Frequency/yr[1]	Unit 2 Frequency/yr[1]
LERF01	Non-SBO with a High Pressure CFE	2.18E-11	2.22E-11
LERF02	Non-SBO with a High Pressure CFE	1.17E-10	1.17E-10
LERF03	Non-SBO with a Low Pressure CFE	2.82E-09	2.82E-09
LERF04	Non-SBO with a TI-SGTR	6.58E-08	6.58E-08
LERF05	Non-SBO with a Low Pressure CFE	5.81E-10	5.81E-10
LERF06	Non-SBO with a PI-SGTR	1.16E-08	1.16E-08
LERF07	Non-SBO with a Low Pressure CFE	1.81E-09	1.81E-09
LERF11	SBO with a High Pressure CFE	4.10E-10	4.10E-10
LERF13	SBO with a High Pressure CFE	3.35E-11	3.35E-11
LERF14	SBO with a Low Pressure CFE	7.38E-10	7.38E-10
LERF15	SBO with a TI-SGTR	3.82E-08	3.82E-08
LERF16	SBO with a Low Pressure CFE	1.55E-10	1.55E-10
LERF17	SBO with a PI-SGTR	1.57E-08	1.57E-08
LERF19	SBO with a Low Pressure CFE	1.25E-10	1.25E-10
CNTMT BYPASS	Associated with SGTR IE or ISLOCA CD Scenarios	1.29E-07	1.29E-07
CNTMT ISO FAILURES	Associated with CD scenarios with containment isolation failures leading to LERF (Includes LERF08, 09, 10, 12, 18 and 20).	5.510E-11	5.510E-11
	Total LERF Release Category Frequency (LERF01 through LERF20)	2.67E-07	2.67E-07

Notes:

1. These values were quantified using a truncation value of 1.00E-14.

Table 5-2 summarizes all of the Level 2 release categories and frequencies. The CDF including uncategorized releases is determined by adding together all Level 2 release categories.

Table 5-2: Comanche Peak Unit 1 and Unit 2 Level 2 Release Categories and Frequencies

Release Category	Definition	Unit 1 Frequency/yr[1]	Unit 2 Frequency/yr[1]
INTACT	Containment Intact	1.39E-06	1.39E-06

SERF	Small Early Release	4.22E-09	4.22E-09
LATE	Late Release	3.78E-06	3.79E-06
LERF	Total Large Early Releases	2.67E-07	2.67E-07
CDF (including uncategorized releases)		5.44E-06[2]	5.45E-06[2]

Notes:

1. These values were quantified using a truncation value of 1.00E-14.
2. This value was calculated as a sum of all release categories (INTACT, SERF, LATE, and LERF).

5.2.3 Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results to reflect the demographics around Comanche Peak Unit 1 and Unit 2. Each of the release categories from Table 5-1 was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 5-3 for reference purposes. Table 5-4 summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 5-3: Summary Accident Progression Bin (APB) Descriptions (Reference 7)

Summary APB Number	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200 psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure at VB or an SGTR, or Alpha mode failures.

Table 5-3: Summary Accident Progression Bin (APB) Descriptions (Reference 7)	
Summary APB Number	Description
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB, No CF Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this reduce bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

For the baseline analysis dose estimates are based on extrapolation of the results of the Surry assessment (Reference 7). For the purpose of sensitivity studies, this analysis uses Comanche Peak doses that were established for the one-time 15 year ILRT extension and are adjusted to account for changes in power level, containment leakage and expected demographics. Population estimates are based on the COL FSAR for proposed Units 3 & 4 through year 2056 representing the most current growth estimates.

Table 5-4: Calculation of Surry Population Dose Risk at 50 Miles (Reference 7)				
Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (person-rem) (4)
1	0.029	0.158	1.23E-07	1.28E+06

2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.012E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.
(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.
(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.
(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.
(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for Comanche Peak Unit 1 and Unit 2.

5.2.4 Population Dose Estimate Methodology

In accordance with Reference 1, the person-rem results in Table 5-4 can be used as an approximation of the dose for Comanche Peak Unit 1 and Unit 2 if it is corrected for allowable containment leak rate (La), reactor power level and the population density surrounding Comanche Peak.

La adjustment:

$$\text{FLeakage} = \frac{\text{La of Comanche Peak Unit 1 and Unit 2 (\%w/o/day)}}{\text{La of reference plant (applicable only to those APBs affected by normal leakage)}}$$

La for Comanche Peak Unit 1 and Unit 2 is 0.1%w/o/day (Reference 26). La for Surry is 0.1%w/o/day.

FLeakage = 0.1 / 0.1

FLeakage = 1

Power level adjustment:

$$\text{FPower} = \frac{\text{Rated power level of Comanche Peak Unit 1 and Unit 2 (MWt)}}{\text{Rated power level of reference plant (MWt)}}$$

The rated power level for Comanche Peak Unit 1 and Unit 2 is 3458 MWt (Reference 26). The rated power level for Surry is 2441MWt.

FPower = 3458 MWt / 2441 MWt

FPower = 1.417

Population density adjustment:

The total population within a 50 mile radius of Comanche Peak Unit 1 and Unit 2 is 2.702E+06. This number is based on the most recent estimates provided in the COL FSAR for proposed Units 3 and 4 through the year 2056 (See Section 5.2). This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Comanche Peak Unit 1 and Unit 2. Note that the numbers reported below may represent a rounded result as displayed in the attached spreadsheets.

Total 2056 estimated Comanche Peak Unit 1 and Unit 2 Population within 50 miles = 2.702E+06

Surry Population within a 50 mile radius from the NUREG/CR-4551 reference plant = 1.23E+06

FPopulation = 2.702E+06/ 1.23E+06 = 2.197

The factors developed above are used to adjust the population dose for the surrogate plant (Surry) for Comanche Peak Unit 1 and Unit 2. For intact containment endstates, the total population dose factor is as follows:

FIntact = FPopulation * FPower * FLeakage

FIntact = 2.197 * 1.417 * 1

FIntact = 3.112

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

FOthers = FPopulation * FPower

FOthers = 2.197 * 1.417

FOthers = 3.112

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. The above adjustments provide an approximation for Comanche Peak Unit 1 and Unit 2 of the population doses associated with each of the release categories from NUREG/CR-4551.

Table 5-5 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Comanche Peak Unit 1 and Unit 2.

Table 5-5: Calculation of Comanche Peak Unit 1 and Unit 2 Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain Comanche Peak Population Dose		Comanche Peak Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06	FOther	3.112	3.98E+06
2	6.46E+05	FOther	3.112	2.01E+06
3	6.46E+05	FOther	3.112	2.01E+06
4	4.95E+05	FOther	3.112	1.54E+06
5	8.12E+05	FOther	3.112	2.53E+06
6	4.23E+02	FIntact	3.112	1.32E+03
7	5.76E+02	FIntact	3.112	1.79E+03

5.2.5 Application of Comanche Peak Unit 1 and Unit 2 PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the Comanche Peak Unit 1 and Unit 2 PRA Level 2 models are not defined in the same terms as reported in NUREG/CR-4551. The Comanche Peak PRA Level 2 model results are defined as four main release categories including INTACT, SERF, LATE, and LERF. In order to use the Level 3 model presented in that document, it was necessary to match the Comanche Peak PRA Level 2 release categories to the collapsed APBs. The Comanche Peak Level 2 release categories and frequencies are from the current simplified Level 2 model (Reference 29). The assignments are shown in Table 5-6, along with the corresponding EPRI classes (see below). The EPRI classes and descriptions are listed in Table 5-7 in addition to the Comanche Peak Level 2 release categories.

Table 5-6: Comanche Peak Unit 1 and Unit 2 Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI Accident Classes

Comanche Peak Level 2 Release Category Frequency	Unit 1 Frequency (per yr)	Unit 2 Frequency (per yr)	Definition	NUREG/CR-4551 APB	EPRI Class
INTACT	1.390E-06	1.390E-06	Containment Intact	6	1
SERF	4.220E-09	4.220E-09	Small Early Release	3	3
LATE	3.780E-06	3.790E-06	Late Release	4	7
LERF01	2.180E-11	2.220E-11	Non-SBO with a High Pressure CFE	2	7
LERF02	1.170E-10	1.170E-10	Non-SBO with a High Pressure CFE	2	7
LERF03	2.820E-09	2.820E-09	Non-SBO with a Low Pressure CFE	3	7
LERF04	6.580E-08	6.580E-08	Non-SBO with a TI-SGTR	5	8
LERF05	5.810E-10	5.810E-10	Non-SBO with a Low Pressure CFE	3	7
LERF06	1.160E-08	1.160E-08	Non-SBO with a PI-SGTR	5	8
LERF07	1.810E-09	1.810E-09	Non-SBO with a Low Pressure CFE	3	7
LERF11	4.100E-10	4.100E-10	SBO with a High Pressure CFE	2	7
LERF13	3.350E-11	3.350E-11	SBO with a High Pressure CFE	2	7
LERF14	7.380E-10	7.380E-10	SBO with a Low Pressure CFE	3	7
LERF15	3.820E-08	3.820E-08	SBO with a TI-SGTR	5	8
LERF16	1.550E-10	1.550E-10	SBO with a Low Pressure CFE	3	7
LERF17	1.570E-08	1.570E-08	SBO with a PI-SGTR	5	8
LERF19	1.250E-10	1.250E-10	SBO with a Low Pressure CFE	3	6
CNTMT BYPASS	1.290E-07	1.290E-07	Associated with SGTR IE or ISLOCA CD Scenarios	5	8
CNTMT ISO FAILURES	5.510E-11	5.510E-11	Associated with CD scenarios with containment isolation failures leading to LERF (Includes LERF 08, 09, 10, 12, 18 and 20).	1	2

5.2.6 Release Category Definitions

Table 5-7 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology (Reference 2). These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 5-7: EPRI Containment Failure Classification (Reference 2)

Class	Description	Comanche Peak Level 2 Release Category Frequency
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant	INTACT
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there are a failure to isolate the containment.	LERF08, LERF09, LERF10, LERF12, LERF18 and LERF20
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.	SERF
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.	N/A
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.	N/A
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.	N/A
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.	LATE, LERF01, LERF02, LERF11, LERF13

Table 5-7: EPRI Containment Failure Classification (Reference 2)

Class	Description	Comanche Peak Level 2 Release Category Frequency
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.	SGTR IE or ISLOCA CD Scenarios

5.3 Impact of Extension on Detection of Component Failures That Lead to Leakage

The ILRT can detect a number of component failures such as liner breach, failure of certain bellow arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 5-7 is divided into two sub-classes, Class 3a and Class 3b, which represent small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance (Reference 20). For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jeffery's non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5 / (217+1) = 0.0023).

In a follow on letter (Reference 17) to their ILRT guidance document (Reference 3), NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for Comanche Peak Unit 1 and Unit 2, as detailed in Section 6 involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large preexisting containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Comanche Peak Unit 1 and Unit 2 Level 2 PRA analyses.

- Class 1 accident sequences may involve availability and or successful operation of containment sprays. It could be assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. However, in this assessment Comanche Peak Unit 1 and Unit 2 do not credit containment spray as a means of reducing releases from Class 3 events.

Consistent with the NEI Guidance (Reference 3), the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten year interval is five years (10 yr/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. An extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak compared to a three year interval.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension (Reference 9)) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis (Reference 5). The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner similar to that at Comanche Peak.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of a containment steel liner. It should be noted that this computation is being applied to provide an upper bound approach to quantify corrosion induced risk. Furthermore, the likelihood of detection of significant corrosion for the 80% of the containment is very high. Regardless, the Calvert Cliffs corrosion likelihood methodology is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the upper containment (cylinder and dome regions in Calvert Cliffs evaluation)
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

5.4.1 Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is conservatively assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-8, Step 1).

- There are two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis. These events have been determined to be applicable at Comanche Peak since due to the similarity between containment liners. The events included in the Calvert Cliffs corrosion assessment process, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the nonvisible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is based on 70 steel-lined containments.

- The Calvert Cliffs analysis used the estimated historical liner flaw probability based on 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date. Since the time of the Calvert Cliffs submittal, three additional relevant liner corrosion events involving concealed corrosion (corrosion initiated on the inaccessible liner surface) were observed and are considered in the corrosion risk assessment. Two of these events occurred at Beaver Valley Unit 1 (References 21 and 22). The third occurred at D.C. Cook Unit 2 (Reference 23). Consistent with the addition of the three observed events, the historical liner flaw probability was established by incrementing the flaw observation time by 12.25 years (September 1996 to July 2014). This re-evaluation resulted in a reduction of the historical liner flaw likelihood to 4.02E-03/year ((2+3) / [70 * (5.5 + 12.25)] = 4.02E-03/year). This value is smaller than the value of 5.2E-03 which is used in the Calvert Cliffs analysis. The conservative value of 5.2E-03 will be used in this Comanche Peak report to remain consistent with the Calvert Cliffs analysis. This approach, while conservative, provides a simplified, direct comparison to the previously evaluated Calvert Cliffs analysis.

- Consistent with the Calvert Cliffs analysis, the steel plate flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel ages. (See Table 5-8, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For Comanche Peak Unit 1 and Unit 2, the ILRT maximum pressure is 50 psig [Reference 29] and ultimate pressure was estimated at 136 psig (Reference 33). Given the above information and consistent with recently approved 15 year ILRT extensions (Reference 30) probabilities of 1% for the shell above the basemat and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4-8, Step 4).

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the upper containment region (See Table 4-8, Step 4).

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 4-8, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

5.4.2 Analysis

Table 5-8: Steel Liner Corrosion Base Case

Step	Description	Upper Containment	Containment Basemat
1	Historical Steel Liner Flaw Likelihood Failure Data: Containment location specific	Events: 2 (Brunswick 2 & North Anna 2) (2)/(70 * 5.5) = **5.2E-03**	Events: 0 (assume half a failure) 0.5/(70 * 5.5) = **1.3E-03**

<table>
<tr><td colspan="6">Table 5-8: Steel Liner Corrosion Base Case</td></tr>
<tr><td>Step</td><td>Description</td><td colspan="2">Upper Containment</td><td colspan="2">Containment Basemat</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Age Adjusted Steel Liner Flaw Likelihood

During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
2.1E-03
5.2E-03
1.4E-02</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
5.0E-04
1.3E-03
3.5E-03</td></tr>
<tr><td colspan="2">15 year average =
6.27E-03</td><td colspan="2">15 year average =
1.57E-03</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years

Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference 5).</td><td colspan="2">0.71% (1 to 3 years)
4.06% (1 to 10 years)
9.40% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td><td colspan="2">0.18% (1 to 3 years)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td></tr>
</table>

Table 5-8: Steel Liner Corrosion Base Case

Step	Description	Upper Containment	Containment Basemat
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	Visual Inspection Detection Failure Likelihood Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	Likelihood of Non-Detected Containment Leakage (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the leakages for the upper containment and the containment basemat as summarized below for Comanche Peak Unit 1 and Unit 2.

Total Likelihood of Non-Detected Containment Leakage Due To Corrosion for Comanche Peak Unit 1 and Unit 2:

At 3 years: 0.00071% + 0.00018% = 0.00089%

At 10 years: 0.0041% + 0.0010% = 0.0051%

At 15 years: 0.0094% + 0.0024% = 0.012%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the three in ten year case is calculated as follows:

- Per Table 5-6, the Comanche Peak Unit 1 and Unit 2 CDF associated with accidents that are not independently LERF or could never result in LERF are Level 2 Release Categories INTACT, SERF and LATE. Therefore the Comanche Peak Unit 1 CDF associated with accidents that are not independently LERF or could never result in LERF is equal to 1.39E-06/yr + 4.22E-09/yr + 3.78E-06/yr = 5.17E-06/yr. The Comanche Peak Unit 2 CDF associated with accidents that are not independently LERF or could never result in LERF is equal to 1.39E-06/yr + 4.22E-09/yr + 3.79E-06/yr = 5.18E-06/yr.

- Per Table 6-3, the EPRI Class 3b frequency is 1.19E-08/yr for Unit 1 and 1.19E-08/yr for Unit 2.

- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 5.17E-06/yr * 0.00089% = 4.61E-11/yr for Unit 1 and 5.18E-06/yr * 0.00089% = 4.61E-11/yr, where 0.00089% was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.

- The three in ten year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 1.19E-08/yr + 4.61E-11/yr = 1.20E-08/yr for Unit 1 and 1.19E-08/yr + 4.61E-11/yr = 1.20E-08/yr for Unit 2.

6 Results

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20), EPRI-TR-104285 (Reference 2) and previous risk assessment submittals on this subject (References 5, 8, 18, 19) have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 6-1 lists these accident classes.

The analysis performed examined Comanche Peak Unit 1 and Unit 2 specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 6-1: Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to Seal-Type B)

Table 6-1: Accident Classes	
Accident Classes (Containment Release Type)	**Description**
5	Small Isolation Failures (Failure to Seal-Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including Very Low and No Release)

6.1 Step 1 - Quantify the Base-Line Risk in Terms of Frequency Per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 6-1 were developed for Comanche Peak Unit 1 and Unit 2 by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 5-6, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion per the methodology described in Section 5.4.

For Unit 1, the total frequency of the categorized sequences is 5.44E-06/yr, the total CDF[2] is 5.44E-06/yr, and the scale factor is 1.0. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency (5.44E-06/yr / 5.44E-06/yr = 1.0). For Unit 2, the total frequency of the categorized sequences is 5.45E-06/yr, the total CDF is 5.45E-06/yr, and the scale factor is 1.0. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency (5.45E-06/yr / 5.45E-06/yr = 1.0).

[2] CDF as established from the summation of the CAFTA Level 2 release classes (see Table 4-2).

This process ensures that the bounding CDF of 5.44E-06/yr for Unit 1 and 5.45E-06/yr for Unit 2 is maintained for the determination of Class 3 states (see below) and effectively distributes the dose impact of the non-represented classes (SERF endstates which are considered Classes 4, 5, and 6) proportionately (per frequencies identified in the adjusted columns of Table 6-2) over the evaluated Classes 1, 2, 7, and 8. The CDF values from Table 4-2 include all release categories (uncategorized results include Classes 4, 5, and 6). Table 6-2 contains the frequencies from the specific categorized sequences and the resulting frequencies due to the application of the scale factor (which redistributes the frequencies to the categorized endstates).

Table 6-2: Comanche Peak Unit 1 and Unit 2 Categorized Accident Classes and Frequencies

EPRI Class	Comanche Peak Release Category	Unit 1 Frequency Based on Categorized Results (per yr)	Unit 1 Adjusted Frequency Using Scale Factor of 1.013 (per yr)	Unit 2 Frequency Based on Categorized Results (per yr)	Unit 2 Adjusted Frequency Using Scale Factor of 1.023 (per yr)
1	Intact Containment (INTACT)	1.390E-06	1.391E-06	1.390E-06	1.391E-06
2	Containment Isolation Failures (LERF08 & LERF17)	5.510E-11	5.514E-11	5.510E-11	5.514E-11
7	Late Containment Failure (LATE) and Containment Failure (LERF01, LERF02, LERF03, LERF05, LERF07, LERF10, LERF11, LERF12, LERF14, LERF16)	3.787E-06	3.790E-06	3.797E-06	3.800E-06
8	Containment Bypass (LERF09 & LERF18) and SGTR (LERF04, LERF06, LERF13 & LERF15)	2.603E-07	2.605E-07	2.603E-07	2.605E-07
Total Frequency		5.37E-06	5.437E-06	5.441E-06	5.447E-06

Class 1 Sequences: This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Containment Intact Level 2 Release Category listed in Table 5-6 minus the EPRI Class 3a and 3b frequency, which are calculated below.

Class 2 Sequences: This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Large Containment Isolation Failures Level 2 Release Category listed in Table 5-6.

Class 3 Sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 [see Section 5.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 [see Section 5.3]

As described in Section 5.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class 3a Frequency (Unit 1) = 0.0092 * (CDF - (Class 2 + Class 8))
= 0.0092 * (5.44E-06/yr – (5.51E-11/yr + 2.61E-07/yr)) = 4.77E-08/yr

Class 3b Frequency (Unit 1) = 0.0023 * (CDF - (Class 2 + Class 8))
=0.0023 * (5.44E-06/yr – (5.51E-11/yr + 2.61E-07/yr)) = 1.19E-08/yr

Class 3a Frequency (Unit 2) = 0.0092 * (CDF - (Class 2 + Class 8))
= 0.0092 * (5.45E-06/yr – (5.51E-11/yr + 2.61E-07/yr)) = 4.78E-08/yr

Class 3b Frequency (Unit 2) = 0.0023 * (CDF - (Class 2 + Class 8))
=0.0023 * (5.45E-06/yr – (5.51E-11/yr + 2.61E-07/yr)) = 1.19E-08/yr

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 100La. These assignments are consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

Note, in the above equations for the Class 3a and 3b release frequencies, the total adjusted release frequency from the appropriate columns of Table 6-2 has been substituted for CDF. As discussed previously this process marginally over-estimates the Class 3 releases.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions. Consistent with guidance provided in EPRI Report No. 1009325, Revision 2-A, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from the Severe Accident Phenomena-Induced Failures Release Category from the Comanche Peak Unit 1 and Unit 2 Level 2 results shown in Table 5-6.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from the Containment Bypass Release Category from the Comanche Peak Unit 1 and Unit 2 Level 2 results shown in Table 5-6.

6.1.1 Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 the NEI Interim Guidance, and guidance provided in EPRI Report No. 1009325, Revision 2-A. Table 6-3 summarizes these accident frequencies by accident class for Comanche Peak Unit 1 and Unit 2.

Table 6-3: Radionuclide Release Frequencies as a Function of Accident Class (Comanche Peak Unit 1 and Unit 2 Base Case)

Accident Classes (Containment Release Type)	Description	Unit 1 Frequency (per Rx-yr)		Unit 2 Frequency (per Rx-yr)	
		Base Case	Base Case Plus Corrosion[1]	Base Case	Base Case Plus Corrosion[1]
1	No Containment Failure	1.331E-06	1.331E-06	1.331E-06	1.331E-06
2	Large Isolation Failures (Failure to Close)	5.514E-11	5.514E-11	5.514E-11	5.514E-11
3a	Small Isolation Failures (liner breach)	4.766E-08	4.766E-08	4.775E-08	4.775E-08
3b	Large Isolation Failures (liner breach)	1.192E-08	1.196E-08	1.194E-08	1.198E-08
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.790E-06	3.790E-06	3.800E-06	3.800E-06
8	Bypass (Interfacing System LOCA)	2.605E-07	2.605E-07	2.605E-07	2.605E-07
CDF	All CET end states	5.441E-06	5.441E-06	5.451E-06	5.451E-06

1. Note that this is based on data developed in Section 5.4. Only Class 3b is impacted by the corrosion.

6.2 Step 2 - Develop Plant Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant specific release analyses were performed to estimate the person-rem doses to the population within a 50 mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 5.2, and summarized in Table 5-5. The results of applying these releases to the EPRI containment failure classification are as follows:

Class 1 = (1.32E+03 person-rem (at 1.0La) + 5.09E+03 person-rem (at 1.0La)) / 2 = 1.55E+03 person-rem (1)

Class 2 = 2.01E+06 person-rem (2)

Class 3a = 1.55E+03 person-rem x 10La = 1.55E+04 person-rem (3)

Class 3b = 1.55E+03 person-rem x 100La = 1.55E+05 person-rem (3)

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 1.54E+06 person-rem (4)

Class 8 = 2.53E+06 person-rem (5)

(1) The derivation is described in Section 5.2 for Comanche Peak Unit 1 and Unit 2. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as an arithmetic average of the dose for these bins and is bounding[3].

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF).

(3) The Class 3a and 3b dose are related to the Class 1 leakage rate as shown. While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100La. Class 3a releases are conservatively assessed at 10La. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) The Class 7 dose is assigned from APB #4 (Late CF)[4].

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from

[3] The use of a simple average dose is bounding as the average over-estimates the proportion of endstates that initiated with a pre-existing isolation failure (APB #7) in the Comanche Peak Level 2 model. These states would be classified as SERFs.

[4] States 3 and 4 map into EPRI release class 7, however, state 3 represents 1% of the contributors to this release class. The dose is selected based on the LATE APB only is conservative as the value does not directly impact the Class 3 doses and is used as an element in the baseline dose for use in fractional dose comparisons.

APB #5 (Bypass).

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology (Reference 2) containment failure classifications, and consistent with the NEI guidance (Reference 1) as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 6-4.

Table 6-4: Comanche Peak Unit 1 and Unit 2 Population Dose Estimates for Population Within 50 Miles

Accident Classes (Containment Release Type)	Description	Unit 1 and Unit 2 Person-Rem (50 miles)
1	No Containment Failure	1.55E+03
2	Large Isolation Failures (Failure to Close)	2.01E+06
3a	Small Isolation Failures (liner breach)	1.55E+04
3b	Large Isolation Failures (liner breach)	1.55E+05
4	Small Isolation Failures (Failure to seal-Type B)	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	1.54E+06
8	Bypass (Interfacing System LOCA)	2.53E+06

The above dose estimates, when combined with the results presented in Table 6-3, yield the Comanche Peak Unit 1 and Unit 2 baseline mean consequence measures for each accident class. These results are presented in Table 6-5 and Table 6-6.

Table 6-5: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Ctmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.33E-06	2.07E-03	1.33E-06	2.07E-03	-7.16E-08
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	4.77E-08	7.41E-04	4.77E-08	7.41E-04	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	1.19E-08	1.85E-03	1.20E-08	1.86E-03	7.16E-06
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-5: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Ctmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.79E-06	5.84E+00	3.79E-06	5.84E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	6.50E+00	5.44E-06	6.50E+00	7.09E-06

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 6-6: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Ctmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.33E-06	2.07E-03	1.33E-06	2.07E-03	-7.17E-08
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	4.78E-08	7.42E-04	4.78E-08	7.42E-04	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	1.19E-08	1.86E-03	1.20E-08	1.86E-03	7.17E-06
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-6: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Ctmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.80E-06	5.85E+00	3.80E-06	5.85E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.45E-06	6.52E+00	5.45E-06	6.52E+00	7.10E-06

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

6.3 Step 3 - Evaluate Risk Impact of Extending Type A Test Interval From 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current ten year value to fifteen years. To do this, an evaluation must first be made of the risk associated with the ten year interval since the base case applies to a three year interval (i.e., a simplified representation of a three in ten interval).

6.3.1 Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is directly impacted. As it is assumed that the new Class 3 endstates arise from previously intact containment states, the intact state frequency is reduced accordingly. The risk contribution is changed based on the NEI guidance as described in Section 5.3 by a factor of 3.33 compared to the base case values. The Unit 1 and Unit 2 results of the calculation for a ten year interval are presented in Table 6-7 and Table 6-8, respectively.

6.3.2 Risk Impact Due to 15-Year Test Interval

The risk contribution for a fifteen year interval is calculated in a manner similar to the ten year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the three year interval value, as described in Section 5.3. The Unit 1 and Unit 2 results for this calculation are presented in Table 6-9 and Table 6-10, respectively.

Table 6-7: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.19E-06	1.85E-03	1.19E-06	1.85E-03	-2.38E-07
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	1.59E-07	2.47E-03	1.59E-07	2.47E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	3.97E-08	6.17E-03	3.98E-08	6.19E-03	2.38E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-7: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.79E-06	5.84E+00	3.79E-06	5.84E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	6.51E+00	5.44E-06	6.51E+00	2.36E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 6-8: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.19E-06	1.85E-03	1.19E-06	1.85E-03	-2.39E-07
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	1.59E-07	2.47E-03	1.59E-07	2.47E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	3.98E-08	6.18E-03	3.99E-08	6.20E-03	2.39E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-8: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.80E-06	5.85E+00	3.80E-06	5.85E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.45E-06	6.52E+00	5.45E-06	6.52E+00	2.36E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 6-9: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.09E-06	1.70E-03	1.09E-06	1.70E-03	-3.58E-07
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	2.38E-07	3.70E-03	2.38E-07	3.70E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	5.96E-08	9.26E-03	5.98E-08	9.30E-03	3.58E-05
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-9: Comanche Peak Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.79E-06	5.84E+00	3.79E-06	5.84E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	6.51E+00	5.44E-06	6.51E+00	3.54E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 6-10: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.55E+03	1.09E-06	1.70E-03	1.09E-06	1.70E-03	-3.59E-07
2	Large Isolation Failures (Failure to Close)	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	0.00E+00
3a	Small Isolation Failures (liner breach)	1.55E+04	2.39E-07	3.71E-03	2.39E-07	3.71E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	1.55E+05	5.97E-08	9.28E-03	5.99E-08	9.31E-03	3.59E-05
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 6-10: Comanche Peak Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	1.54E+06	3.80E-06	5.85E+00	3.80E-06	5.85E+00	0.00E+00
8	Bypass (Interfacing System LOCA)	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	0.00E+00
CDF	All CET end states	N/A	5.45E-06	6.53E+00	5.45E-06	6.53E+00	3.55E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

6.4 Step 4 - Determine the Change in Risk in Terms of LERF

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Comanche Peak Unit 1 and Unit 2, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). For Unit 1, the baseline LERF based on a test frequency of three times in ten years is 1.20E-08/yr as shown in Table 6-11. Based on a ten year test interval from Table 6-7, the Class 3b frequency (conservatively including corrosion) is 3.98E-08/yr; and, based on a fifteen year test interval from Table 6-9, it is 5.98E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three to fifteen years to one in fifteen years for Unit 1 is 4.78E-08/yr as shown in Table 6-11. Similarly, the increase due to increasing the interval from ten to fifteen years is 2.00E-08/yr as shown in Table 6-11. For Unit 2, the baseline LERF based on a test frequency of three times in ten years is 1.20E-08/yr as shown in Table 6-12. Based on a ten year test interval from Table 6-8, the Class 3b frequency (conservatively including corrosion) is 3.99E-08/yr; and, based on a fifteen year test interval from Table 6-10, it is 5.99E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three to fifteen years to one in fifteen years for Unit 2 is 4.79E-08/yr as shown in Table 6-12. Similarly, the increase due to increasing the interval from ten to fifteen years is 2.00E-08/yr as shown in Table 6-12.

As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF for Comanche Peak Unit 1 and Unit 2 is below the threshold criteria for a very small change when comparing both the fifteen year results to the current ten year requirement, and the fifteen year results compared to the original three year requirement. See Table 6-11 and Table 6-12 for more information.

6.5 Step 5 – Determine the Impact on the CCFP

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI Report No. 1009325, Revision 2-A. The NRC has previously accepted similar calculations (Reference 9) as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The list below shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects. Note that the numbers used are rounded to the second decimal place and the final values represent the numbers calculated in the attached spreadsheets.

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

$CCFP_{3,\ Unit\ 1}$ = [1 - (1.33E-06/yr + 4.47E-08/yr) / 5.44E-06/yr] * 100% = 74.65%

$CCFP_{3,\ Unit\ 1}$ = 74.65%

$CCFP_{10,\ Unit\ 1}$ = [1 - (1.19E-06/yr + 1.59E-07/yr) / 5.44E-06/yr] * 100% = 75.17%

$CCFP_{10,\ Unit\ 1}$ = 75.17%

$CCFP_{15,\ Unit\ 1}$ = [1 - (1.09E-06/yr + 2.38E-07/yr) / 5.44E-06/yr] * 100% = 75.53%

$CCFP_{15,\ Unit\ 1}$ = 75.53%

$\Delta CCFP_{15\text{-}3\ Unit\ 1} = CCFP_{15,\ Unit\ 1} - CCFP_{3,\ Unit\ 1}$ = 0.88%

$\Delta CCFP_{15\text{-}10\ Unit\ 1} = CCFP_{15,\ Unit\ 1} - CCFP_{10,\ Unit\ 1}$ = 0.37%

$\Delta CCFP_{10\text{-}3\ Unit\ 1} = CCFP_{10,\ Unit\ 1} - CCFP_{3,\ Unit\ 1}$ = 0.51%

$CCFP_{3,\ Unit\ 2}$ = [1 - (1.33E-06/yr + 4.78E-08/yr) / 5.45E-06/yr] * 100% = 74.70%

$CCFP_{3,\ Unit\ 2}$ = 74.70%

$CCFP_{10,\ Unit\ 2}$ = [1 - (1.19E-06/yr + 1.59E-07/yr) / 5.45E-06/yr] * 100% = 75.21%

$CCFP_{10,\ Unit\ 2} = 75.21\%$

$CCFP_{15,\ Unit\ 2} = [1 - (1.09E\text{-}06/yr + 2.39E\text{-}07/yr) / 5.45E\text{-}06/yr] * 100\% = 75.58\%$

$CCFP_{15,\ Unit\ 2} = 75.58\%$

$\Delta CCFP_{15\text{-}3\ Unit\ 2} = CCFP_{15,\ Unit\ 2} - CCFP_{3,\ Unit\ 2} = 0.88\%$

$\Delta CCFP_{15\text{-}10\ Unit\ 2} = CCFP_{15,\ Unit\ 2} - CCFP_{10,\ Unit\ 2} = 0.37\%$

$\Delta CCFP_{10\text{-}3\ Unit\ 2} = CCFP_{10,\ Unit\ 2} - CCFP_{3,\ Unit\ 2} = 0.51\%$

The change in CCFP of approximately 0.88% for Unit 1 and 0.88% for Unit 2 by extending the test interval to fifteen years from the original three in ten year requirement is judged to be very small (i.e. less than 1.5% per the EPRI submittal guidance).

6.6 Summary of Results

The results from this ILRT extension risk assessment for Comanche Peak Unit 1 and Unit 2 are summarized in Table 6-11 and Table 6-12.

Table 6-11: Comanche Peak Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	1.55E+03	1.33E-06	2.07E-03	1.19E-06	1.85E-03	1.09E-06	1.70E-03
2	2.01E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	5.51E-11	1.11E-04
3a	1.55E+04	4.77E-08	7.41E-04	1.59E-07	2.47E-03	2.38E-07	3.70E-03
3b	1.55E+05	1.20E-08	1.86E-03	3.98E-08	6.19E-03	5.98E-08	9.30E-03
7	1.54E+06	3.79E-06	5.84E+00	3.79E-06	5.84E+00	3.79E-06	5.84E+00
8	2.53E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	2.60E-07	6.58E-01
Total	N/A	5.44E-06	6.50E+00	5.44E-06	6.51E+00	5.44E-06	6.51E+00
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		2.60E-03		8.66E-03		1.30E-02	
Delta Total Dose Rate[1]	From 3 yr	N/A		5.84E-03		1.00E-02	
	From 10 yr						
% change in dose rate from base	From 3 yr	N/A		N/A		4.19E-03	
	From 10 yr	N/A		0.09%		0.15%	
3b Frequency (LERF) Per-Rem/Yr		1.20E-08		3.98E-08		5.98E-08	
Delta LERF	From 3 yr	N/A		2.79E-08		4.78E-08	
	From 10 yr	N/A		N/A		2.00E-08	
CCFP %		74.65%		75.17%		75.53%	
Delta CCFP %	From 3 yr	N/A		0.51%		0.88%	
	From 10 yr	N/A		N/A		0.37%	

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

Table 6-12: Comanche Peak Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)							
EPRI Class	**DOSE Per-Rem**	**Base Case 3 in 10 Years**		**Extend to 1 in 10 Years**		**Extend to 1 in 15 Years**	
		CDF/Yr	**Per-Rem/Yr**	**CDF/Yr**	**Per-Rem/Yr**	**CDF/Yr**	**Per-Rem/Yr**
1	4.41E+03	1.33E-06	2.07E-03	1.19E-06	1.85E-03	1.09E-06	1.70E-03
2	1.14E+06	5.51E-11	1.11E-04	5.51E-11	1.11E-04	5.51E-11	1.11E-04
3a	4.41E+04	4.78E-08	7.42E-04	1.59E-07	2.47E-03	2.39E-07	3.71E-03
3b	4.41E+05	1.20E-08	1.86E-03	3.99E-08	6.20E-03	5.99E-08	9.31E-03
7	8.75E+05	3.80E-06	5.85E+00	3.80E-06	5.85E+00	3.80E-06	5.85E+00
8	1.43E+06	2.60E-07	6.58E-01	2.60E-07	6.58E-01	2.60E-07	6.58E-01
Total	N/A	5.45E-06	6.52E+00	5.45E-06	6.52E+00	5.45E-06	6.53E+00
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		2.61E-03		8.68E-03		1.30E-02	
Delta Total Dose Rate[1]	From 3 yr	N/A		5.85E-03		1.00E-02	
	From 10 yr						
% change in dose rate from base	From 3 yr	N/A		N/A		4.20E-03	
	From 10 yr	N/A		0.09%		0.15%	
3b Frequency (LERF) Per-Rem/Yr		1.20E-08		3.99E-08		5.99E-08	
Delta LERF	From 3 yr	N/A		2.79E-08		4.79E-08	
	From 10 yr	N/A		N/A		2.00E-08	
CCFP %		74.70%		75.21%		75.58%	
Delta CCFP %	From 3 yr	N/A		0.51%		0.88%	
	From 10 yr	N/A		N/A		0.37%	

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

7 Sensitivities

7.1 Sensitivity to Corrosion Impact Assumptions

The Comanche Peak Unit 1 and Unit 2 results in Table 6-5 through Table 6-10 show that including corrosion effects calculated using the assumptions described in Section 5.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the upper containment and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 7-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 4.59E-12/yr for Unit 1 and 4.60E-12/yr for Unit 2. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 7-1: Steel Plate Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Unit 1 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 Years (per Rx-yr)		Unit 2 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 Years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion	Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Upper Containment, 0.1% Basemat)	Base Case (10% Upper Containment, 100% Basemat)	4.78E-08	1.84E-10	4.79E-08	1.85E-10
Doubles every 2 yrs	Base	Base	4.80E-08	3.28E-10	4.81E-08	3.29E-10
Doubles every 10 yrs	Base	Base	4.77E-08	5.31E-11	4.78E-08	5.32E-11
Base	Base	15%	4.79E-08	2.58E-10	4.80E-08	2.58E-10
Base	Base	5%	4.78E-08	1.11E-10	4.79E-08	1.11E-10
Base	10% Upper Containment, 1% Basemat	Base	4.95E-08	1.84E-09	4.96E-08	1.85E-09
Base	0.1% Upper Containment, 0.01% Basemat	Base	4.77E-08	1.84E-11	4.78E-08	1.85E-11
Lower Bound						
Doubles every 10 yrs	0.1% Upper Containment, 0.01% Basemat	5% Upper Containment, 1% Basemat	4.77E-08	3.19E-15	4.78E-08	3.19E-15
Upper Bound						
Doubles every 2 yrs	10% Upper Containment, 1% Basemat	15% Upper Containment, 100% Basemat	4.77E-08	4.59E-12	4.78E-08	4.60E-12

7.2 Sensitivity to Class 3B Contribution to LERF

For Unit 1, the Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 1.20E-08/yr (Table 6-5). Extending the interval to one in ten years results in a frequency of 3.98E-08/yr (Table 6-7). Extending it to one in fifteen years results in a frequency of 5.98E-08/yr (Table 6-9), which is an increase of 4.78E-08/yr from three in ten years to once in fifteen years.

For Unit 2, the Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 1.20E-08/yr (Table 6-6). Extending the interval to one in ten years results in a frequency of 3.99E-08/yr (Table 6-8). Extending it to one in fifteen years results in a frequency of 5.99E-08/yr (Table 6-10), which is an increase of 4.79E-08/yr from three in ten years to once in fifteen years.

If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF for Unit 1 and Unit 2 due to extending the interval from three in ten to one in fifteen is below the RG 1.174 threshold for very small changes in LERF of 1.00E-07/yr.

7.3 Potential Impact From External Events Contribution

The latest information related to external events for Comanche Peak Unit 1 and Unit 2 is from the Extended Power Uprate submittals. These submittals included information which was extended from the Individual Plant Examination for External Events (IPEEE) (Reference 24) submittals. The external events considered included fire, seismic, high winds, external flooding, and nearby facility and transportation accidents.

<u>Fire Assessment</u>

Early assessments of fire risk vulnerabilities were performed as part of the IPEEE. Comanche Peak analyses utilized the EPRI Five Induced Vulnerability Evaluation (FIVE) process. Comanche Peak is currently performing a Fire-PRA to update the risk associated with fire. Until those results are available the ILRT guidance provides a method to estimate a fire LERF by using the most recent internal events CDF to LERF ratio. Using that methodology a bounding fire induced CDF was calculated as 2.09E-05 which did not exceed 1.0E-04 per year.

Seismic Assessment

The seismic assessment implemented a Seismic Margin Methodology (SMM) that is based on the EPRI methodology described in EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin, (Revision 1)". This Methodology consisted of defining the equipment required to safely shutdown the plant following a review level seismic event and then evaluating the equipment through walkdown and margin analysis to show that the equipment will in fact survive at the review level seismic accelerations. Since CPSES is categorized as a reduced scope plant, the review level earthquake is the same as the design basis for the Seismic Category I systems, structures and components, namely the Safe Shutdown Earthquake (SSE).

For a reduced scope plant, the NRC specified that the review level earthquake should be the SSE ground response spectra and in-structure response spectra. The scope of the seismic margin evaluation includes the following important considerations: 1) since the review level earthquake (RLE) is the SSE, all components that are designed to SSE levels are assumed to be acceptable at the RLE, and 2) no seismic margin evaluations above the SSE are required. Thus, the seismic margin evaluation for the reduced scope plant consists of two principle tasks: first, to demonstrate the seismic design of Safe Shutdown Equipment List (SSEL) equipment at the SSE level and second, to perform field review/walkdowns of the equipment.

To accomplish the first task, the Seismic Review Team (SRT) conducted a detailed review of the design documentation and verified the seismic design bases and seismic pedigree of SSEL equipment. The second task involved a detailed field review in which the SRT reviewed the important attributes of the seismic equipment, as described in Appendix A of EPRI NP-6041, with particular emphasis being put on anchorage of equipment and systems interaction. A similar review was done of containment systems. The results of the IPEEE seismic margin evaluation demonstrate that there are no vulnerabilities from seismic events at CPNPP. These results of the IPEEE were later confirmed to still hold true upon review of recent seismic hazard data (Reference 27). Furthermore, Comanche Peak submitted a Seismic Hazard and Screening Report (Reference 32) that demonstrates ground motion response spectra (GMRS) well below the SSE.

Based on the above assessment, the seismic risk at CPNPP is not expected to be a significant contributor to CDF or LERF and has little impact on the results of the ILRT extension assessment. As such, for the purposes of this analysis, the seismic contribution will be shown as "N/A" in Table 7-2 and will not contribute to the overall external events CDF.

High Winds Assessment

An evaluation of high winds, external floods and transportation events and other hazards were performed during the IPEEE. The analysis included the development of a tornado hazard model using the reported tornado events, in the statistical data base from the National Severe Storms Forecast for an area surrounding CPNPP plant site. Based on reviews of CPNPP tornado design criteria and detailed plant walkdowns, component and structural vulnerabilities were identified. Fragilities of these vulnerable structures were developed and integrated into a plant risk model, which was derived from the accident sequence models developed for analysis of internal events as part of CPNPP Individual Plant Examination.

The overall core damage frequency due to tornadoes at CPNPP was estimated to be 3.70E-06 per year in the IPEEE (Reference 24). It should be noted that the high winds analysis has not yet been updated to the current ASME PRA Standard requirements. It is expected that the updated high winds CDF will in fact be smaller than the IPEEE value similar to the reduction seen in the internal event CDF. Still, the overall results indicate that the core damage risk from a tornado strike at CPNPP is quite low. The dominant sequences did not involve tornado-induced failures of plant structures or equipment; rather they involve tornado-induced loss of offsite power. This is due to the fact that nearly all risk-significant equipment is protected within Seismic Category I structures which are designed to withstand tornadoes up to the design basis tornado. These results demonstrate that there is no plant-specific vulnerability at CPSES from high winds. The results of the IPEEE were later reconfirmed to still hold true upon review of recent wind hazard data (Reference 31).

Floods, Transportation & Nearby Facility Accidents and Others:

An evaluation of external floods and transportation events and other hazards were performed during the IPEEE. That assessment concluded that; the category I building structures are not threatened from external flooding, even at the worst conditions of probable maximum precipitation or potential dam failures. The potential maximum oil leak has very little chance of affecting the safety related structures of the station. In the case of a gas line break, the concentration of gas at any plant air intake is well below the lower flammability limit. The incident heat energy from a postulated gas well explosion is substantially well below the solar insolation. The land routes around the station are far away from the plant proper, and are lightly traveled as to pose any type of hazard for CPNPP. In addition, the probability of an aircraft impact on the station is estimated to be 1.19E-9 or one crash in 8.4 million years. The toxic chemicals used inside the plant are under strict procedural controls which should ensure that normal plant operation is not endangered in any manner. Also, the onsite or off-site storage/transportation of chlorine is not expected to pose any hazard.

The area surrounding the station was reviewed for other plant-specific external events that may affect the safety of the plant. With the exception of natural gas exploration, no industrial growth can be expected to occur in the site vicinity. The only hazardous materials (excluding local gas stations and materials not directly related to CPNPP) regularly manufactured, stored, used, or transported in the site vicinity are crude oil and natural gas transported through the pipelines. This review did not identify any further possible external events which might pose any threat to the plant.

These evaluations included a review of the external event hazards at the plant and the licensing basis, an assessment of whether there have been any significant changes since the IPEEE was issued. The results indicated there are no significant changes to the site since the IPEEE was issued that would alter the assessment of these hazard impacts. As these hazards were screen during the IPEEE, no core damage or LERF frequencies were developed at that time.

External Events Summary

Table 7-2 below lists the Comanche Peak CDF values for each external event type that are used to determine the potential impact from the External Events contribution.

Table 7-2: Comanche Peak External Events Summary

External Event Type	Comanche Peak Unit 1 CDF/yr	Comanche Peak Unit 2 CDF/yr
Internal Events *	4.08E-06	4.08E-06
Seismic	N/A	N/A
Other Hazards (High Winds /External Floods/Transportation)**	3.70E-06	3.70E-06
Fire **	2.09E-05	2.09E-05
Total	2.87E-05	2.87E-05

* - Current internal events values for CPNPP Units 1 and 2 (Reference 28)

** - External events values taken from CPNPP Units 1 and 2 IPEEE results (Reference 24)

Combining the External Events CDF values and the Internal Events CDF yields a CDF estimate of 3.10E-05/yr (Unit 1) and 3.11E-05/yr (Unit 2). As the above information is based on a combination of current and IPEEE results, no conclusions should be drawn on the relative contribution of events.

The change in LERF from extending the Type A test interval can be conservatively estimated using the total CDF values to determine the external event contribution. These CDF values were specifically used to determine the Class 3b frequency based on the external events contribution. The factors for determining the increase in the non-detection probability of a leak described in Section 5.3 were applied to the Class 3b base value frequencies to determine the 3b frequencies for the once per ten year test and once per fifteen year test for each unit. Note that the numbers used are rounded to the second decimal place and the final values represent the numbers calculated in the attached spreadsheets.

Class 3b Frequency (three per ten year test) = 0.0023 * (CDF – (Class 2 + Class 8))

Class 3b Frequency (Unit 1) = 0.0023 * (3.10E-05/yr – (5.51E-11 + 2.61E-07) = 7.07-08/yr

Class 3b Frequency (Unit 1) (once per ten year test) = 3.33 * 7.07E-08/yr = 2.36E-07/yr

Class 3b Frequency (Unit 1) (once per fifteen year test) = 5.00 * 7.07E-08/yr = 3.54E-07/yr

Class 3b Frequency (Unit 2) = 0.0023 * (3.11E-05/yr – (5.51E-011 + 2.60E-07)) = 7.09E-08/yr

Class 3b Frequency (Unit 2) (once per ten year test) = 3.33 * 7.09E-08/yr = 2.36E-07/yr

Class 3b Frequency (Unit 2) (once per fifteen year test) = 5.00 * 7.09E-08/yr = 3.55E-07/yr

Table 7-3 shows the results of these calculations.

Table 7-3: Comanche Peak Estimated Total LERF Including External Events Impact

Case	3b Frequency (3 per 10 year test) per year	3b Frequency (1 per 10 year test) per year	3b Frequency (1 per 15 year test) per year	LERF Increase (3 per 10 to 1 per 15 year test) per year
Unit 1 Internal Events Contribution (From Base Case Table 6-11)	1.20E-08	3.98E-08	5.98E-08	4.78E-08
Unit 1 Total Contribution including External Events	7.07E-08	2.36E-07	3.54E-07	2.83E-07
Unit 2 Internal Events Contribution (From Base Case Table 6-12)	1.20E-08	3.99E-08	5.99E-08	4.79E-08
Unit 2 Total Contribution including External Events	7.10E-08	2.36E-07	3.55E-07	2.84E-07

Using the above approach results in a total LERF (Class 3b) value of 3.54E-07/yr for a permanent once per 15 year ILRT program for Unit 1 and 3.55E-07/yr for Unit 2. These frequencies remain below the Regulatory Guide 1.174 criteria of 1.00E-05/yr following the ILRT extension. Furthermore, the increase in total LERF from the three per ten year test to the once per fifteen year test is 2.83E-07/yr for Unit 1 and 2.84E-07/yr for Unit 2, both of which are within the range of the Regulatory Guide 1.174 criteria of 1.00E-07/yr to 1.00E-06/yr for a small change in risk with total LERF remaining less than 1.00E-05/yr.

8 Conclusions

The risk impact of permanently extending the Type A ILRT test frequency to once in fifteen years is as follows:

- The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 4.78E-08/yr for Unit 1 and 4.79E-08/yr for Unit 2. As such, the estimated change in LERF for Unit 1 and Unit 2 is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 (Reference 4) also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from External Events was also made. In this case, the total class 3b contribution to LERF including External Events was conservatively estimated as 2.83E-07/yr for Comanche Peak Unit 1 and 2.84E-07/yr for Comanche Peak Unit 2. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfies both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.00E-02 person-rem/yr for Unit 1 and 1.00E-02 person-rem/yr for Unit 2. Note that this value is based on internal events only and does not consider external events. The total population dose is thus increased to 6.51 person-rem/yr for Unit 1 and 6.53 person-rem/yr for Unit 2. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose (6.51E-02 for Unit 1 and 6.53E-02 for Unit 2), whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 25). Moreover, the risk impact when compared to other severe accident risks is negligible. Note that CPNPP is below both criteria for meeting the definition of a very small population dose.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen year interval is 0.88% for Unit 1 and 0.90% for Unit 2. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 25). Both Unit 1 and Unit 2 prove to be below 1.5 percentage points and thus are considered to be very small.

Therefore, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the Comanche Peak Unit 1 and Unit 2 risk profile.

ATTACHMENT 7 TO TXX-15001
CPNPP PRA TECHNICAL ADEQUACY

Probabilistic Risk Assessment Technical Adequacy

1.0 CPNPP PRA Model Description

The CPNPP PRA model that will be used for this application is a Level 1 and Level 2 analysis of Internal Events, including Internal Flood, for At-Power operation. An ASME PRA Standard compliant Fire PRA is in progress.

2.0 CPNPP PRA Model Background

The ASME PRA Standard (Reference B), as endorsed by Regulatory Guide 1.200 Revision 2 (Reference A), was used to demonstrate the technical adequacy of the CPNPP PRA model used for this application. The Peer Review of the CPNPP Revision 4 model is the baseline for PRA Standard compliance at CPNPP. PRA Standard compliance for PRA model updates and applications subsequent to Revision 4 is programmatically required at CPNPP.

The CPNPP PRA Model of Record (MOR) and documentation is in full compliance with the internal events portion of the ASME PRA Standard and Regulatory Guide 1.200. The MOR addresses Level 1 and Level 2 analysis of Internal Events, including Internal Flood, for At-Power operation. The model had a PWROG full scope Peer Review in March 2011. The Peer Review was performed against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (Reference B) and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Regulatory Guide (RG) 1.200 Revision 2 (Reference A). Further, the Peer Review was performed using the process defined in Nuclear Energy Institute (NEI) 05-04 (Reference C). Immediately following the Peer Review, the model was revised to incorporate model and documentation changes in response to the Peer Review Findings & Observations (F&O) and issued as Revision 4A.

A minor periodic update was performed to bring the current Model of Record (MOR) for Comanche Peak Nuclear Power Plant to Revision 4B.

In order to provide a more complete description of PRA technical adequacy, some specific elements of the historical PRA model warrant additional discussion. The principal historical issues are described below and have been fully addressed with the Peer Review and subsequent model Revision 4A. These are: 1) Recovery of Faulted Equipment, 2) LOOP Non-Recovery Probabilities, 3) Human Reliability Analysis, 4) Dependency Analysis, and 5) Data Analysis.

Recovery of Faulted Equipment – Recovery of faulted equipment is no longer credited in the CPNPP PRA model.

LOOP Non-Recovery Probabilities – Loss of Offsite Power (LOOP) non-recovery probabilities are now based on a lognormal rather than Weibull curve fit in the convolution integrals. This results in comparable non-recovery probabilities at shorter times and higher non-recovery probabilities at longer times after the LOOP event. In the current non-recovery analysis, database recovery values are limited such that there are no probabilities less than 1E-02.

Human Reliability Analysis – A comprehensive update of the Human Reliability Analysis was done as part of the Peer Review model update. The current Human Reliability Analysis is based on the EPRI HRA Calculator methodology. In all cases, screening values have been replaced. Detailed evaluations of human errors based on plant procedures and operating practices were done and fully documented in the HRA Calculator.

Dependency Analysis – A detailed dependency analysis was performed as part of the recent update. The current dependency analysis is comprehensive and follows the current industry guidance provided in the EPRI HRA Calculator methodology.

Data Analysis – A comprehensive data analysis was done as part of the Peer Review model update. NUREG/CR-6928 data was used to establish priors for a Bayesian update. CPNPP Units 1 and 2 plant specific data were used to calculate the posterior probabilities.

3.0 Technical Adequacy of the CPNPP PRA Model

Model Represents the As-built and As-operated Plant

Luminant employs a programmatic approach to establish and maintain the technical adequacy and plant fidelity of the PRA models. This approach includes both a detailed PRA maintenance and update process and the use of self-assessments and independent peer reviews. This approach ensures that the PRA continues to adequately represent the as-built, as-operated plant.

In addition, requirements are established for controlling the model and associated computer files including documentation of the PRA model and basis documents and electronic storage of PRA update information, PRA models, and PRA applications.

The CPNPP PRA model was developed using programmatic controls to help assure that the model reflected the as-built and as-operated Plant. This process included gathering detailed as-built and as-operated plant information and operating plant data, discussions with system engineers and operators, and plant walk down. The continuing PRA maintenance and update process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. These processes are defined in the governing procedure ECE 2.15 (Reference D), and subordinate implementation guidelines. The procedure and guidelines define the processes for implementing regularly scheduled and interim PRA model updates, and for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, software, industry operating experience, etc.)

Plant Changes Not Yet Incorporated into the PRA Model

As of the date of the submittal of this LAR, there are no plant changes of significance that have not been incorporated into the PRA model; however, it is anticipated that from time to time, some additional plant changes will occur that ought to be reflected in the model. To this end, a PRA model update tracking database is used to identify and track plant changes that could impact the PRA model. A review of open items in the tracking database has been performed and confirmed that there are no plant changes of significance that have not been incorporated that would impact the results of the analysis.

Consistency with Applicable PRA Standards

A full scope Peer Review of the CPNPP PRA Model Revision 4 was completed by the PWROG in March 2011 against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (Reference B) and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200 (Reference A). The outcome of the Peer Review showed that the CPNPP MOR 4 meets ASME/ANS RA-Sa-2009 Parts 1, 2, and 3 Capability Category II or better for nearly all of the Supporting Requirements. After Findings and Observations were addressed through post-Peer Review model work and documentation (MOR 4A), all but three Supporting Requirements meet Capability Category II or better. The F&O Findings and their dispositions, including the three Category I exceptions, are provided in Table 1 below. The MOR 4A was used and found to be technically adequate to support the implementation of the Surveillance Frequency Control Program at CPNPP.

Summary of the Risk Assessment Methodology

The following information is covered in the body of the LAR and is not repeated here.

- The parts of the PRA used to support the application and how these are implemented in the PRA model, including a definition of the acceptance criteria used for this application.
- The scope of risk contributors addressed by the PRA model. This includes the scope of the PRA model (i.e. internal and external), identifying appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
- Summary of the risk assessment methodology used to assess the risk of the application, include how the PRA model was modified to appropriately model the risk impact of the change request.

Identification of Key Assumptions and Approximations

For the CPNPP PRA MOR, key assumptions and approximations were identified and documented in the various notebooks. In addition, modeling uncertainties associated with scope or level of detail (modeling choices) for the baseline PRA are documented and validated in the respective notebooks; for example, the individual system models were analyzed with respect to the assumptions documented in the system notebooks to understand the impacts of those assumptions on the overall model. Finally, a comprehensive uncertainty analysis was done using a consensus methodology developed by EPRI (Reference G). Each area of uncertainty is generally related to certain key assumptions and approximations associated with the model and each was characterized through sensitivity studies. The sensitivity studies provide a mechanism for meeting the ASME high level requirements and provide a better understanding of the model that will ultimately be used in the decision-making process supporting risk informed applications.

Resolution of Peer Review Findings and Observations

The results of the PWROG Peer Review of the CPNPP PRA Model Revision 4 (Reference E) showed 21 Findings and 55 Observations / Suggestions and 4 Best Practices. All findings were resolved by either modifying the model, enhancing the documentation, or a combination of both. The suggested resolutions provided by the peer review team for each F&O were considered and generally incorporated. However, in some cases the PRA staff felt a different solution was best for CPNPP and still satisfied the F&O as written.

After the F&O's were fully addressed through post-Peer Review model work and documentation, as reflected in Revision 4A, all Supporting Requirements previously judged to have been "Not Met" were judged to be Cat II or greater.

Four of the seven Supporting Requirements previously judged to have been "Cat I" were judged to be Cat II or greater. The following three SR's remain at Cat I:

- LE-C11 (no F&O)
- IFEV-A6 (no F&O)
- IFSN-A6 (F&O 6-4)

All but the following three suggestions have been incorporated into the model and/or documentation. The following suggestions are being carried in the update database:

F&O	Impact Item
1-18	28
2-3	29
2-20	30

CPNPP has submitted and received a SER for the NEI Option 5b Surveillance Frequency extension program. The NEI Option 5b application is similar to this submittal in that they both deal with surveillance frequency extensions. Excerpts from the NRC SER for CPNPP pertaining to the CPNPP PRA quality are provided in Attachment 1 and the overall conclusions are considered applicable to this submittal.

Identify Parts of the PRA that Conform to Capability Categories Lower Required for the Application

The current model of record, revision 4B meets Category II or better for all but 3 SR requirements. These remaining SR's were judged by the Peer Review Team to meet "Cat I". In Section 3.2.4.1 of the SER for EPRI TR-1 009325, Revision 2, the NRC staff states that Capability Category I of American Society of Mechanical Engineers (AS ME) PRA standard shall be applied as the standard for assessing PRA quality for IRLT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies. Therefore, the CPNPP PRA model is considered to be adequate to support the evaluation of changes to ILRT frequencies.

4.0 External Events Considerations

External hazards were evaluated in the CPNPP Individual Plant Examination for External Events (IPEEE) report which was submitted in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4). The results of the CPNPP IPEEE are documented in the CPNPP IPEEE Main Report (Reference F).

Luminant does not yet have quantifiable models for external hazards that meet the requirements of the ASME / ANS combined standard. A Fire PRA using the guidance in the ASME PRA Standard is in progress. A High Winds and Other Hazards PRA is planned to start following completion of the Fire PRA. Given that CPNPP's updated Ground Motion Response Spectrum (GMRS) is well below the SSE at all frequencies, seismic risk at the site is extremely unlikely to be a significant issue for any risk-informed application. The updated GMRS has allowed CPNPP to be one of the few plants that will submit a minimal Seismic risk evaluation in response to the 10CFR50.54f letter that was issued following the Fukushima Accident. Therefore, a Seismic PRA is not planned for CPNPP.

5.0 Summary

The CPNPP PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk informed applications.

6.0 References:

A. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, US Nuclear Regulatory Commission, January 2007.

B. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, February 2009.

C. NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Nuclear Energy Institute, November 2008.

D. ECE 2.15, *"Risk and Reliability Functions,"* Comanche Peak Nuclear Power Plant Procedure.

E. Letter: McCoy, D.E. to Zachariah, T. dated May 19, 2011 SUBJ: RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Comanche Peak Nuclear Power Plant Probabilistic Risk Assessment, Westinghouse LTR-RAM-II-11-038, Attachment: RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Comanche Peak Nuclear Power Plant Probabilistic Risk Assessment, Westinghouse Proprietary Class 2.

F. CPNPP IPEEE Report: ER-EA-008, "Individual Plant Examination of External Events for severe Accident Vulnerabilities – CPSES" dated June 1995. [Docketed as TU Electric Letter logged TXX-95171 from C.L. Terry to USNRC dated June 27, 1995]

G. EPRI TR1016737. "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments", December 2008

Attachment 1 – Excerpts from NRC SER for CPNPP Risk Informed Applications Option 5b Submittal

From 5b SER - ADAMS Accession No. ML12067A244

A proposed amendment would adopt the NRC-approved Technical Specifications Task Force (TSTF) traveler TSTF--425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b" (Reference 4). TSTF--425, Revision 3, would relocate the frequencies of most periodic surveillances from the TS to a new licensee-controlled program, the Surveillance Frequency Control Program (SFCP), and would impose requirements for the new SFCP in the Administrative Controls section of the TSs

3.2.1.4.1 Quality of the PRA

The quality of the CPNPP PRA must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the quality of the PRA.

RG 1.200 is the NRC's developed regulatory guidance for assessing the technical adequacy of a PRA. Revision 2 of this RG endorses (with comments and qualifications) the use of the American Society of Mechanical Engineers (ASME) I American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 10), NEI 00-02, "PRA Peer Review Process Guidelines" (Reference 11), and NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 12). Revision 1, of this RG had endorsed the internal events PRA standard ASME RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 13). For the internal events PRA, there are no significant technical differences in the standard requirements; therefore, assessments using the previously endorsed internal events standard are acceptable.

The licensee has performed an assessment of the PRA models used to support the SFCP using the guidance of RG 1.200 to assure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability category II is required by NEI 04-10 for the internal events PRA, and any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including by the use of sensitivity studies where appropriate.

The CPNPP PRA internal events model identified as Revision 4 was subject to a full scope industry peer review by the Pressurized Water Reactor Owners' Group (PWROG) in March 2011, using the internal events PRA standard endorsed by RG 1.200, Revision 2. The current model of record, identified as Revision 4A, includes responses to identified findings and observations (F&O) from the peer review, which were summarized in Table 2-1, "F&O Summary Findings," of the licensee's submittal

(Reference 1). In addition, the licensee identified in Table 2-2, "SRs Assessed as Not Met or Category I for the CPNPP PRA," of its submittal (Reference 1) the supporting requirements (SRs) from the internal event PRA standard which were identified as not met or only meeting capability category I.

Based on the licensee's assessment using the applicable PRA standard and RG 1.200, the NRC staff concludes that the level of PRA quality, combined with the proposed evaluation and disposition of remaining gaps to capability category II of the standard, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177.

Table 1

F&O Summary Findings

F&O Number	Associated Supporting Requirement	F&O Details	Resolution
1-7 (including Suggestion 6-5)	IFSN-A16 IFSN-A14	SW flood sources in the diesel generator rooms (1-084-SW and 1-085-SW), and as a result the areas themselves, were screened in part based on the availability of alarms indicating a pipe failure and the ability to isolate the break before the SW system would be lost resulting in an initiating event. However, credit for the operator isolation is not noted as part of the basis for screening the source and area in R&R-PN-021 Table 4.5-2.	In the original analysis, it was determined that a loss of a single Service Water (SW) train would cause a Technical Specification (TS) immediate plant shutdown due to the loss of an Charging pump. It was determined that there were viable operator actions to isolate the diesel generator SW without affecting the Charging pumps. After further consultation with plant licensed operators it was later concluded that the loss of a Charging pump function did not result in an immediate TS plant shutdown. The use of operator actions to screen these flood scenarios therefore was not necessary and the scenarios were required to be screened by other criteria. These rooms are now currently screened by the criteria that they do not cause an initiating event. Table 4.5-2 of PN-021 was revised to state that a loss of one emergency diesel generator and a single train of service water (SW) do not cause an immediate plant shutdown. The SW pipe break for the scenarios in question is assumed to occur in the diesel room. The flood in the diesel room will propagate outside the safeguards building and not cause a plant trip. No further analysis was conducted on these specific operator actions in question.

		Assessment: Cat I is MET	Cat II or better based on this resolution
1-10	HR-E3 HR-E4 HR-G5 HR-I1	Documentation of past operator interviews was provided. The manner in which these interviews were performed is not documented so it is not clear that detailed talkthroughs were performed and in any case this information is from the 2003 time frame. This information was supplemented in the latest revision with specific questions to operations personnel that are documented in R&R-PN-020 Attachment 4. However, the documentation of the operator interviews is not judged by the review team to be sufficient to support peer review and model updates.	Additional Operator interviews were performed as follows: The modeled Human Interactions fall into three general categories of response: 1) simple alarm response, 2) plant trip response using EOP/EOS procedures (i.e. typical response), and 3) response following loss of function using FR / ECA procedures. Several HIs from each category were selected as representative of the category. Standard briefing sheets and open ended response areas were prepared with the goal of confirming the response models (including timing) for modeled scenarios and that the PRA analyst's interpretation of procedures was consistent with plant observations and training procedures. The briefing packages were used to capture the interview observations of three Operations Unit Supervisors (i.e. 3 crews). For each modeled action, the Unit Supervisors stepped through the associated procedures, including timing estimates and crew dynamics where appropriate. The Operations Support

		Assessment: Cat II or better MET	Supervisor (current SRO) also provided "EOPs for Engineers" training for the PRA analysts. This training covered EOP usage and operations protocols including training standards, timing standards, etc. The Operations Support Supervisor also provided response and timing information for a number of specific modeled actions. The results from these interviews were consistent with the modeled HFEs and did not require significant changes to any HEPs. Cat II or better MET
1-11	HR-G7 HR-H3 QU-C2	The top 7 HFE combinations appearing in the quantification results were reviewed. Three had incorrect ordering of the independent failures (i.e., the wrong HFE in the combination was assigned as the independent failure) and one had an incorrect assignment of complete dependence. An example of a combinations with the incorrect assignment of the independent failure is TLXHICOMB106. In this combination, the first event should be TLXHISGPSLLY based on review of the event tree. However, TLXHICOND45Y is treated as the independent event in calculating the combined HEP. The correct sequencing of the events would lead to a different outcome for the joint probability.	The HRA dependency analysis was completely revised. An updated version of the HRA Calculator® allowed intervening successes and local delay timings to be adjusted so as to correctly assign the independent failure. All HFEs were reviewed to insure that a consistent definition of T_0 was used and that cues were appropriate for the accident sequence context where the combination appears. As stated previously, delay times for individual actions within a combination were locally adjusted when necessary to provide the correct ordering of actions for the dependency analysis. All combinations were reviewed to confirm that the correct HFE had been designated as the independent event. All combinations were reviewed to verify that intervening successes had been

		In addition, one of the reviewed combinations, revealed that complete dependence was assigned for actions with an intervening success. For example, HFEs TLXHIHPR13SY, Failure to Align High Pressure Recirculation, and TLXHIEOS13SY, Failure to align Low Pressure Recirculation are assigned complete dependence (TLXHICOMB111). However, in the context of the sequence, there is an intervening success in this sequence (Secondary Depressurization) which would result in the HFEs being assessed as independent. Assessment: HR-G7 was NOT MET	properly identified. Cat II or better based on this resolution
1-12	QU-D4 LE-F2	R&R-PN-035 Section 5.8 compares the total LERF for CPNPP with several other Large Dry Containment 4-loop PWRs. However, there is no comparison at the level of significant contributors or plant damage states. Without the comparison of contributor information, it is not really possible to determine how similar the LERF results are to other plants and whether excessive conservatisms have skewed the results. For example, the contribution to LERF from early containment failure is significantly higher than usually found for large dry containments. This may be valid for CPNPP and based on some plant-specific design	A comparison of the LERF results to plants of similar design at the significant contributor and PDS levels was added to R&R-PN-035 and RXE-LA-CPX/0-105. This comparison shows that the CPNPP LERF results are reasonable based on plant specific features and thermal hydraulic analysis. Cat II or better based on this resolution

		feature, but it does not appear that there was consideration of the possibility that this is driven by modeling assumptions rather than design. Assessment: LE-F2 was NOT MET	
1-16	HR-I3	R&R-PN-020 Section 3 is titled "Assumptions and Sources of Uncertainty." However, those assumptions that are sources of uncertainty are not clearly identified. Assessment: Cat I - III MET	PN-020, §3.0 has been subdivided into 3 subsections dealing with modeling choices, assumptions, and source of uncertainty. The sources of uncertainty have been verified to be characterized under QU-F4 as described in PN-041. Cat I - III MET
2-8	SC-A6	For SGTR, there appears to be no consideration of the case where an MSSV opens following the SGTR (not as a result of overfill) and sticks open, allowing the SG to depressurize. Assessment: Cat I - III MET	Additional plant specific thermal-hydraulic analysis performed for SGTR case with stuck open MSSV. No changes to success criteria or model logic were necessary. Cat I - III MET
2-12	AS-A4 AS-A5 AS-A7 AS-A10 AS-C2	The model uses a simple, two sequence event tree for all transient groups. This is not fundamentally a problem, since it is possible to build the event-specific plant response into the sequence top logic. However, in order to do that, the event-specific progression needs to be discussed in detail, the possible sequences defined, and each possibility either qualitatively argued away as a non-contributor or implemented in the logic model. This has not been done. Unlike the non-transient initiators, the format of the	The transient initiating event group discussion in R&R-PN-013 has been divided into sub-groups based on EOP progression. Each sub-group section discusses specific progression, timing, system states, procedures used, and operator actions. The sub-sections have been formatted similar to the other initiating events and include an ERG Actions portion that is specific to the sub-group. Model logic has been re-verified to confirm that no possible sequences have been excluded. Additional analysis has verified that failure to isolate

		discussion for the transient initiators is that a single detailed discussion is provided for a general progression. This is followed by brief discussions of some of the other initiating event groups, but not in sufficient detail such that the progression can be clearly understood. The MSLB discussion touches on the qualitative basis for not addressing failure to isolate the MSLB, but there is insufficient justification and it does not affect the actual impact of single and multiple SG blowdown (for example, is make-up required to compensate for primary shrinkage to prevent drawing a bubble into the RCS piping). This is finally followed by a single discussion of the ERG actions relevant to transients, but again this is only general in nature and does not present specific information on the different actions and procedures that are used for the various initiating event groups and how those could impact both plant and operator response (which would necessitate inclusion in the top logic). Assessment: AS-C2 was NOT MET; Assessment: AS-A10 was Cat I	single or multiple steamlines following a MSLB will not uncover the core, thus not requiring success criteria different from the overall transient group. Cat II or better based on this resolution
2-13	AS-A7 AS-C3	Key sources of model uncertainty and assumptions related to the accident sequence modeling are	1. Additional discussion of offsite power recovery modeling and sequence development added to PN-039, App. E as follows: "A consideration of the

		documented in R&R-PN-013, Section 3. There are some sources of uncertainty that are missing. For example: 1. The way offsite power recovery is handled in the accident sequences is not discussed as a potential source of uncertainty. The model assumes that, once offsite power is recovered the sequence is over. Therefore, the actual recovery and operation of the mitigating systems after power recovery may introduce unique failures that are not addressed in LOOP sequences without SBO. In particular, after recovery of offsite power many things have to be done manually that would occur automatically for LOOP without SBO, and some equipment will be in a different state (i.e., handswitches in pull-to-lock). 2. While WCAP-15831 may be considered a "consensus" ATWS model, the WCAP includes consideration of ATWS from power levels less than 40% (States 1 and 2) that are not addressed in the CPNPP model. While these may be lesser contributors to the ATWS risk (~10%), the omission of parts of the "consensus" model does constitute a potential source of uncertainty that needs to	off-site power recovery scheme is successful recovery and what happens once power is recovered. First, the methodology applies a non-recovery probability to the LOOP initiating event. That is, the cutset containing this initiating event has a probability associated with failure to recover power within a defined time frame. This failure leads to core damage. Second, if power recovery is successful, the additional failures required to lead to core damage produce cutsets that are non-minimal to the cutset with failed power recovery or another cutset, which would thus be subsumed from the cutset file. The logic allows for the LOOP initiator to propagate through the model with the on-site AC power available. This generates cutsets that consider many of the additional failures associated with restarting and loading equipment or additional manual actions that may be required. For example, the following two cutsets are generated by the current methodology. Cutset one: LOOP IE, CCF of both EDGs, non-recovery probability Cutset two: LOOP IE, TDAFW FTS, CCF of MDAFW, CCP A FTS, PORV B FTO With no AFW and 1 PORV failed to open, success criteria will require 2 CCPs If off-site power was successfully recovered in cutset number one

		be addressed. In addition, WCAP-15831 Section 8.2 states that "ATWS events can be initiated from a wide range of initiating events. The ATWS analysis for Westinghouse PWRs (Reference 6) established that the limiting events, with regard to RCS peak pressure, are the loss of load with subsequent loss of all MFW and complete loss of normal feedwater. These limiting events are both assumed to be initiated from normal operation at full power." It is further stated in Section 8.2.1 that "The model presented in this section assumes MFW is lost for all anticipated transient events. If MFW continues to operate, then the event does not need to address the pressure relief response, including AFW and AMSAC, but only requires long-term shutdown. A split that accounts for MFW continuing to operate may be added to plant specific ATWS models if desired." it is not clear that the modeling for the LOMFW top event captures all potential losses of MFW following the initiating event. For example, flood events INIT-F0-AUXSWA and INIT-F0-AUXSWB, as analyzed, would trip the CW pumps	and the scenario continued, the cutset would look like: Cutset three: LOOP IE, CCF of both EDGs, power recovered, TDAFW FTS, CCF of MDAFW, CCP A FTS, PORV B FTO The quantification software would look at cutset three and cutset two and identify that cutset three was non-minimal to cutset two and remove it before the results would be written to the cutset file. Thirdly, if the model logic was modified to account for the successful restoration of off-site power, cutsets may be generated that contain unique failures after recovery that does not apply before recovery. That is, that after recovery of offsite power many actions may have to be done manually that would occur automatically for LOOP without SBO, and some equipment will be in a different state. For example, 1) Loop IE, CCF of Both EDGs, Operator fails to establish FW after recovery of power, 2) Loop IE, CCF of Both EDGs, CCF of normal power tie breakers to safety busses to close or 3) Loop IE, CCF of Both EDGs, Operator fails to properly sequence loading of busses. For these remaining cases, the impact on CDF would be insignificant. This can be seen by looking at the overall make-up of these non-generated cutsets. For CPNPP the LOOP initiating event frequency is approximately

		due to actuation of the flood sensing switches in the condenser pits resulting in a loss of condenser coincident with the reactor trip. However, these are treated as transients with MFW available in the ATWS analysis. In addition, random failure of MFW following reactor trip is not addressed in the fault tree logic for top event LOMFW (this was included in the Braidwood model described in Section 9.1 of the WCAP). Therefore, it is not clear that the CPNPP modeling is entirely consistent with the "consensus" model and the potential uncertainty introduced by the deviations is not discussed.	E-2, the CCF of the EDGs is approximately E-4, providing an E-6 starting point probability. A best estimate of successful LOOP recovery prior to core damage would lower these cutsets by at least an order of magnitude, resulting in an E-7 cutset probability. For non-operator failure based scenarios, because of redundancy and/or diversity of equipment/success paths, at least two additional failures would have to occur in order to cause core damage. This would provide at least an E-4 failure probability. For operator failure based scenarios, given that once off-site power is restored, the focus of the staff would be on re-energizing the safety busses, followed by restoration of mitigating equipment and systems. Therefore a failure probability of E-3 would be an appropriate value based on current similar HRA analyses. This would put the non-generated cutsets in the E-10 to E-11 range (or lower) for a given core damage scenario. Given the current model CDF value (~3E-06) and the contribution of LOOP (~ 10 to 15 percent) to CDF, these scenarios would not significantly contribute to overall CDF." The above information has been added to the Quantification Support File notebook, R&R-PN-039. 2. The ATWS event tree has been revised to pass all anticipated transient events discussed in the WCAP (i.e. no loop, no ISI) through the

			LOMFW logic. The split accounting for MFW continuing to operate has been removed. Concerning LOOP, the WCAP further states (§5.4): "Since the impacts on CDF and RCS integrity from LOSP/AWTS events are very small, this event will not be important to the plant risk profile or to risk-informed decision process for assessing changes to a plant." Regarding states 1 and 5 (low power), WCAP §5.4 also states: "Since the CDF and the impact on CDF are dominated by ATWS state 3 / 4 this state is the most important one to consider in plant specific PRA models. The other modes of operation are small contributors to plant risk and will not be important to the plant specific risk profile or to the risk-informed decision process for assessing changes to a plant." The results of the WCAP show that states 1 and 5 contribute less than 2.5% to ATWS risk. Since ATWS risk at CPNPP is a 0.1% contributor, the potential contribution to overall CDF risk from ATWS states 1 and 5 is on the order of 0.0025%. The uncertainty due to exclusion of ATWS states 1 and 5 is therefore confirmed to be insignificant to plant specific risk profile or to the risk-informed decision process for assessing changes to the plant. The above information has been added to the Accident Sequence notebook, R&R-PN-013.

		Assessment: AS-A7 was NOT MET	Cat II or better based on this resolution
2-16	HR-C2 HR-D2 HR-E2 HR-F2 HR-G2	R&R-PN-020 Section 4.1.2 states that "In general, caution was exercised when considering both an error of omission (EOM) and an error of commission (EOC) for the same activity. For most component manipulations, these activities were judged to be mutually exclusive. For example in the case of a repositioning a valve following a test, an error of omission skips the reposition. This would be a reasonable error. An error or commission, however would be to reposition the valve, i.e. the desired outcome, and is not considered." This is insufficient basis for excluding EOC. EOC could include (for this example) "repositioning" the wrong valve (correct intent, wrong action). This same thought process was applied to the EOC for post-initiator actions and was also not	All HFEs have been re-analyzed to include appropriate Errors of Commission. Cat II or better based on this resolution

		adequately justified. Assessment: HR-C2 was NOT MET	
2-18	HR-H3 QU-C1	The process followed for dependency analysis utilizes the HRA Calculator® to identify the combinations of HFEs that appear in cutsets. The process used a quantification run with a truncation level of 1E-14 to identify the HFE combinations to assessed, but used nominal HEP values so it cannot be assured that all important combinations were identified. Assessment: QU-C1 was NOT MET	The cut set used in the re-analysis of dependency was generated by setting all HEPs to 1E-01 and re-quantifying at 1E-12, (which meets the ASME PRA Standard for setting a truncation value). Additional combinations were obtained and appropriately analyzed for dependency. 1E-01 is generally at least two orders of magnitude higher than the HEP values and is sufficiently elevated to identify important combinations. Cat II or better based on this resolution
3-1	IE-C5 IE-D2	From a methodology point of view, and per report R&R-PN-008A, Rev 4, with the exception of the LOOP initiators, a reactor year basis and an appropriate availability factor was used. So, it is deemed that the analysis meets the CC-I/II as a whole. However, because LOOP, as stated in section 4.7 of R&R-PN-008A, Rev 4, uses a calendar year basis instead of a reactor year, an F&O was generated to document the need to convert the LOOP initiating events to reactor year based frequencies. Assessment: IE-C5 was Cat I/II	LOOP IE frequencies were adjusted to a reactor year basis and all other IE frequencies were re-verified to be calculated on a reactor year basis and documented in R&R-PN-008A. Cat II or better based on this resolution
4-1	IE-A1 IE-A4 IE-A5	In general, the initiating event analysis seems to have identified a representative set of initiating events. However,	1. The Initiating Event Analysis (PN-003) was revised to document the system-by-system initiating event review used to

	IE-A7 IE-B2 IE-C3 IE-C11 IE-D2	the following areas were identified where the documentation was missing or deficient in the current revision: 1. Appendix D of R&R-PN-008A (Rev. 3A) documents a systematic evaluation of each system to identify potential system initiating events. R&R-PN-024 contains the support system initiators that include SW, CCW, CH and switchyard. It seems the systematic evaluation was performed, but not documented in detail in Revision 4 of R&R-PN-003 or R&R-PN-008A. (IE-A1, IE-A5, IE-B2) 2. Page 20 of R&R-PN-008A (Attachment 5) contains a summary of the plant-specific initiating event experience. However, the treatment of events resulting in an unplanned controlled shutdown that includes a scram prior to reaching low-power conditions is not discussed. (IE-A7) 3. Section 8.0 of R&R-PN-003 refers to a review of Licensee Event Reports (LERs), covering the period from September of 1988 through May of 1998, to identify any industry initiating events which could not be placed in one of the identified Initiating Event categories.	identify potential system initiating events. In addition, PN-003 was also updated to incorporate the documentation of the IE-D2 supporting requirement elements. 2. Added following text to PN-008A, §4.0: "A review of recent (see §4.2) plant operating experience was performed to identify occurrences of initiating events since the previous update. The only screening criterion used in this review was that a plant trip would not be counted if it was a planned event as part of a planned shutdown for refueling. In addition to at-power events, the review also looked for shutdown events that could also occur at power and events occurring during an unplanned controlled shutdown that resulted in a trip prior to reaching low power conditions." 3. NUREG/CR-6928 provides a reasonable expectation of common initiators for PWRs. A few of these initiators are not applicable to CPNPP. Similarly, CPNPP has a small number of "unique" initiators that have been added due to analysis or plant experience. Attachment 1 of PN-008A contains the mapping between IE's from industry sources and the CPNPP PRA model. PN-008A, §4.0 has the following summary: "Development of the initial CPNPP PRA model included a comprehensive search for initiating events and was documented in R&R-PN-003. Additionally, 2411 NRC LERs

		There is no documentation in the notebook providing details of this review such that it can be independently verified or reconsidered during future model updates. (IE-A4) 4. Recovery through cross-tie of the Unit 1 and Unit 2 SW and CCW systems (SWXTIE and CCWXTIE respectively) is credited in the Support System Initiating Event Fault Trees. However, this is not documented in R&R-PN-003 or R&R-PN-024. (IE-C3, IE-C11)	from September 1988 to May 1998 were re-reviewed during the revision to PN-003. All events could be placed within one of the existing initiating event categories. This search process is not repeated for PRA updates since the general set of PWR initiators is well established. A general search of recent industry events (INPO Operational Transients database and Ref. 2.7) did not identify any previously unseen types of initiating events. Review of references 2.1, 2.17, 2.18, 2.19 did not identify any initiators that are not included in the model, or any precursors that would indicate potential initiators were overlooked. Existing initiating event groups are consistent with other United States PWRs and do not require modification for this update. The initiator list bounds plant experience. The model freeze date for this update is 6/30/08. Attachment 1 is a summary of the updated internal initiating event frequencies. The calculations shown in Attachments 3-7 are documented in Excel spreadsheet "Rev4_Initiating_Events.xls". Calculations are performed as instructed in Ref. 2.5. A review of recent (see §4.2) plant operating experience was performed to identify occurrences of initiating events since the previous update. The only screening criterion used in this review was that a plant trip

		Assessment: IE-D2 was NOT MET	would not be counted if it was a planned event as part of a planned shutdown for refueling. In addition to at-power events, the review also looked for shutdown events that could also occur at power and events occurring during an unplanned controlled shutdown that resulted in a trip prior to reaching low power conditions. This review identified 7 plant trips during the period under consideration. These trips are listed in Attachment 5. No events were screened out during this review, nor were any initiating events identified that are not included in the current model. The system engineers were also interviewed to determine if the system models were missing any potential indicators. None were identified. The interviews are documented in ref. 2.42." Similarly, Operations reviewed the initiating event list during the updated operator interview. No changes were identified. 4. All SW and CCW crosstie recovery credit has been removed from the Support System Initiating Event fault trees. Cat II or better based on this resolution
4-4 Suggestion	IE-A8 IE-D2	The following reviews for identification of potential initiating events Interviews with plant personnel were not performed to determine if potential initiators have been	Following text added to §4.0 of PN-008A: "The system engineers were interviewed to determine if the system models were missing any potential indicators. None were identified. The interviews are documented

		overlooked. Assessment: IE-A8 was Cat I	in R&R-PN-008A. Several Operations shift supervisors reviewed the list of current initiating events to determine if any potential initiating events had been overlooked. None were identified." Cat II or better based on this resolution
4-12	DA-D4 DA-E2	Section 4.3 of R&R-PN-008 states that the resulting posterior distributions were reviewed and "any inconsistencies examined by comparing them to prior and plant experience. Results were determined to be reasonable based on the weight of evidence." However, there is no documentation associated with this review. Assessment: Cat I - III MET	Review results from the comparison of the ratio derived from plant specific data with prior mean values has been added to R&R-PN-008. Cat I - III MET
4-13	SY-B1 DA-D6	Section 4.5 of R&R-PN-008 states that a review of industry data sources and relative risk importance for SYSIMP groups supported deletion of eight common cause groups. The deleted CCF "component types" were fans, dampers, air compressors, bistables and non-safety batteries. Assessment: Cat II or better MET	The statement on review of common cause groups was incorrectly interpreted to indicate screening or exclusion. This statement in R&R-PN-008 has been clarified as follows: "To support the definition of common cause groups, component types were reviewed against industry data sources and relative risk importance for SYSIMP groups. No CCF events associated with significant basic events were excluded in the definition of common cause groups." Cat II or better MET
4-14	DA-D3	The CPNPP PRA includes mean values and statistical	The Multiple Greek Letter method for estimating CCF

	DA-E2	representations of the uncertainty intervals for the parameter estimates. However, the uncertainty parameters for the CCF events are not included. Assessment: Cat III MET	mean values is a method adopted from NUREG 5485 which is cited as a source in the ASME standard. Though this method does not readily support statistical representation of uncertainty intervals, other sources of uncertainty have been considered. As noted in Appendix D.5 of NUREG 5485, "the uncertainties due to judgment required in interpretation and classification of failure events and the assessment of impact vectors are the most significant of all sources of uncertainty." This discussion of uncertainty for CCF parameter estimation has been added to R&R-PN-008. The data notebook includes an explicit reference to R&R-PN-041, Uncertainty Analysis, which addresses EPRI recommendations for treatment of uncertainty. Cat III MET
4-15	DA-C4 DA-D1	The CPNPP PRA includes many SSCs with plant-specific parameter estimates (see Attachment 3 of R&R-PN-008). However, there is no documented systematic process or criteria to determine which SSCs should be evaluated for the plant-specific estimates, including the potentially significant basic events.	A systematic review of plant specific data identified those components with sufficient, relevant plant data. All components with sufficient data were updated to generate plant specific parameter estimates. Data sources reviewed for changes in failures or failure modes included Maintenance Rule, Mitigation System Performance Indicator (MSPI), EPIX and consultations with System and Component Engineers. This discussion of the update

		Assessment: Cat II or better MET	process and criteria was added to R&R-PN-008. Cat II or better MET
4-24	IE-B5 IE-C3 IE-C11 QU-D1	Sections 5.2.1, 5.2.2, 5.2.3, 5.2.4 of R&R-PN-022 discuss the cutset reviews performed for CPNPP PRA. Issues were identified with two specific cutsets that require additional discussion and/or justification: 1. Cutset #9 in the CDF results contains two events (RHACHCOOL & RHBCHCOOL) which represent the conditional probability a RH train will fail upon loss of the essential chilled water that provide the room cooling. Each event has the conditional probability of 0.688 based on the RXE-SY-CP1/1-028 (1992). It is not clear whether this conditional probability is justifiable. In addition, it seems the RHACHCOOL & RHBCHCOOL events should be based on a joint probability when these two events show together in a cutset. (QU-D1) 2. Cutset #10 contains SWXTIE that credits Unit 2 SW system upon Loss of SW system in Unit 1 followed by an induced RCP Seal LOCA which would result in a start signal for the EDGs. It is not clear whether the operators have enough	1. The treatment of room cooling was reviewed and determined to be correctly applied, i.e. individual room heat loads are different, thus different probabilities are reasonable. However, due to the uncertainty regarding potential dependency, cutsets containing failures of both trains are treated as completely dependent. A replacement event equal to the highest probability of the pair is substituted in place of the independent events. A sensitivity case [R&R-PN-041] has been performed to address the uncertainty of this assumption. Further discussion of this topic is addressed in App. D of R&R-PN-039. 2. All SW and CCW crosstie recovery credit has been removed from the Support System Initiating Event fault trees. PN-024 discusses credit for cross-ties to mitigate core damage but not in determining the SSIE frequencies. This crosstie function is credited as a recovery only with both trains in the other unit available. As modeled, one train from the other unit cannot be used to supply both units. Use of the crosstie does not prevent inducing an RCP Seal LOCA; nor does it prevent operators from taking required actions (e.g. stopping the EDGs on a

		time to make the crosstie in time to provide the cooling to EDGs on Unit 1 before the diesels fail. (IE-B5, IE-C3, IE-C11, QU-D1) Assessment: QU-D1 was NOT MET	loss of SW) prior to alignment of the crosstie. Cat II or better based on this resolution
4-29 Suggestion	QU-A2 QU-D6 QU-F3	Section 6.0 of R&R-PN-022 provides the discussions of the significant contributors to CDF, the initiator contributions, and top event contributors for each event tree. Section 7.0 of R&R-PN-022 provides significant contributors from CCF events, operator actions and independent events. However, the sequence level contributors are not indentified in the notebook. Assessment: QU-F3 was Cat I	Discussion of significant sequences has been added to PN-022. Cat II based on this resolution
4-31	QU-E4 QU-F4	R&R-PN-041 provides the results of uncertainty and sensitivity results, and other PRA notebooks identify the potential sources of model uncertainty. However, it is not clear how these sources of uncertainty affect the PRA model. Assessment: QU-F4 was NOT MET	R&R-PN-041 Section 5.1 describes the application of the EPRI approach to CPNPP. Cat II based on this resolution
4-34	QU-F6	No documentation was found in R&R-PN-022, 39 and 41 providing a quantitative definition of significant basic event, cutset, and accident sequence.	The quantitative definition for significant basic event, significant cutset, and significant accident sequence is as described in ASME/ANS PRA Standard, part 2. This definition has been explicitly added to PN-022, 039,

		Assessment: QU-F6 was NOT MET	and 041. Cat II based on this resolution
4-35	IE-C8 IE-C10 DA-C16	The CCW, SW, and CH initiating event fault trees use a MTTR factor, which is calculated based on the data from the Maintenance Rule database. It is not clear whether the data screening was appropriately handled for the initiating event criteria. In addition, the MTTR factor is applied using rules based recovery rather than being explicitly modeled in the SSIE fault trees as required by this SR.	The data used to calculate the MTTR value were screened to identify unavailability events not associated with planned test and maintenance. Detailed data includes dates, durations and the reason for unavailability. A table detailing screen results was added to R&R PN-008. MTTR events are explicitly included in SSIE models. The following is included in PN-024, §4.1: "Each SSIE tree has been developed such that every train is modeled with the operating equipment relying on an annualized exposure time. At an appropriate location, where the trains meet in the logic, an event representing the mean time to repair of the redundant train was placed. This event effectively replaces the annualized value of the redundant equipment with the MTTR exposure time. In this way, the logic will always result in a yearly frequency at the top while any of the operating trains may represent the initial annualized failure. An example of this approach is shown below, and each MTTR event used in the model is subsequently discussed. Standby equipment with relevant failure modes and common-cause failure events are modeled with an MTTR of 24 hours so that the mitigating logic may be used directly in most

		Assessment: IE-C10 was NOT MET	cases (see Section 3.2). These events, therefore, are modeled such that they bypass the additional MTTR events. (Common-cause failures are discussed in Section 6.0.) Other dual-train failure modes (such as shared tank ruptures) receive the yearly exposure which propagates to the top without further manipulation. Cat II based on this resolution
6-4	IFSN-A6	RG 1.200 Revision 2 documents a qualified acceptance of this SR. The NRC resolution states that to meet Capability Category II, the impacts of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement) must be qualitatively assessed using conservative assumptions. Assessment: IFSN-A6 was Cat I based on the qualification in RG 1.200, Revision 2	As noted, the qualitative analysis of impingement, pipe whip, humidity, and condensation concerns was not conducted for the PRA flood model; CPNPP previously completed a design basis High Energy Line Break (HELB) calculation. Since most of the high energy systems are located in compartments that are segregated from the rest of the plant by watertight doors and have flood paths directly to the plant yard there should be minimal impact to PRA equipment. A qualitative analysis of these conditions will be performed on as needed basis. Assessment: remains at Cat I
6-7	IFQU-A6	The human actions taken from the main control room during flooding scenarios (listed in	The nineteen HFE's that appear in flood cutsets were re-reviewed. Eleven of these HIs

		Table 4.9-1) were judged to not incur additional stress above the same actions when analyzed for the internal events analysis. This judgment is based on: (1) the components and cues not being affected by the flood, (2) the actions being based on steps as defined in a procedure, (3) the operators being highly trained in executing the procedure steps (many are memorized as immediate actions), and (4) the actions being backed up by supervision in the control room during the flooding scenario. Additionally, most of the actions are assumed to be taken early in a sequence before the determination that a flood is occurring (an average time frame of 10 minutes is assumed). Thus, few of the actions are expected to be taken in the long term as a scenario progresses. Certainly, the components in the control room (those physically manipulated by the operators) should not be affected by the flood scenarios. The lack of impact on cues is not certain. All the actions appear to be based on procedural guidance for which the operators are trained. Control room supervision is expected. However, the assumption regarding the actions being taken early within a scenario does not apply to several of the main control room actions that	are performed in the control room and are simple actions (e.g. start an alternate pump, stop a pump, etc) in response to an alarm or EOP. The judgment is that these actions are not impacted by flood scenarios because they are simple, occur shortly after the trip, and are within the EOP. We strongly believe that the Operators will stay within the ERG network, as trained, until they have stabilized the plant. Further these events occur early enough that stress levels are judged to be unchanged from the level originally assessed for the event. WOG ERGs are symptom based. That is, the operators respond to plant indications rather than performing diagnosis of the event. No specific time limit is applied to this review. One HFE concerns equipment failed due to flooding. The HFE is therefore N/A. The remaining 7 actions have some portion of the response performed remotely (i.e. in the field). The general process was to increase transit time where the flood could cause the PEO to stop and have additional discussion with the control room and/or re-route to the destination. In lieu of more specific information, the transit time was doubled. Changing timing in the HRA methodology may change the dependency of recovery actions, with potential subsequent increase in the Human Error Probability (HEP). This approach was judged reasonable since the actions that were modified already include

		are listed in Table 4.9-1. At least nine actions from Table 4.9-1 were identified which could reasonably be expected to be performed in a time frame beyond the assumed, average 10 minutes. Among these actions were: - The failure to establish feed-and-bleed within 25 (or 30) minutes - The failure to align for low pressure recirculation (up to 41.8 minutes) - The failure to align for hot leg recirculation (up to 270 minutes) - The failure to start a standby SW pump (up to 37 minutes) - The failure to depressurize and begin RHR SDC (up to 1433 minutes) Assessment: Cat I - III MET	the effects of high stress in the base analysis. Three of the seven are not impacted by flood scenarios. TLXHICSTFILY and AFXHICSTFILY are "refill the CST" actions and do not occur for at least 5.5 hours after plant trip. In this case the flood impacts are assumed to be terminated. EPXHICHASW_Y does not occur in a flood area. Four of the seven remote actions were judged to be potentially impacted by the flood. The potential impact was judged to likely be an increase in transit time due to additional communication with the control room or the need to take an alternate path. For these cases, transit time was doubled. Where appropriate, dependency levels were changed as a result of increased timing. In two of the four events the dependency level did not change, thus the HEP did not change. In the remaining two events, the dependency level increased, thus increasing the associated HEP. Cat I - III MET
No F&O included	IFEV-A6	R&R-PN-021 Section 4.7 indicates that the flood initiating event frequencies were based on the EPRI 1021086 failure data combined with plant-specific piping	During the internal flooding (IF) analysis a search for previous IF events at CPNPP was performed and none were found. A Bayesian update with no specific plant events would incur a non-

		lengths. No Bayesian updating with plant-specific operating experience or adjustment based on engineering judgment was performed. Assessment: Cat I is MET	conservative result; therefore no Bayesian update was performed as there is no impact to application evaluations. Assessment: remains at Cat I
No F&O included	LE-C11	No credit was taken for continued operation of equipment after containment failure. RXE-LA-CPX/0-105 Table 6-1 specifically notes that "No credit is taken for operation of the ECCS/CS system after containment failure or for operator actions or other equipment that could be impacted by containment failure because there are none that are significant." It is not clear that this is equivalent to justifying "any credit given" as required for CC II/III. Assessment: Cat I is MET	Since no credit has been taken for continued operation after containment failure, justification cannot be provided. Impact on specific applications will be evaluated as needed. Assessment: remains at Cat I
2-7 (Suggestion F&O)	SC-A6	The MAAP calculations (RXE-LA-CPX/0-103 and RXE-LA-CPX/0-104) are generally consistent with features and procedures. However, the requirement for switchover to hot leg recirculation is conservative, and may impact the CDF results and, thus, the insights on dominant contributors.	To address the conservatism, Hot Leg Recirculation requirements were removed for Small and Very Small LOCA based on Westinghouse ERG documents and comparison with other plants.