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Washington, DC 20555-0001

Subject: License Amendment Request for a One-Time Extension Of Containment Integrated Leakage Rate Test Interval.
Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

The purpose of this letter is for Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90 to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). The proposed license amendment would revise TS section 5.5.12 "Primary Containment Leakage Rate Testing Program" to allow a one-time extension of the 10 CFR 50 Appendix J, Type A, Integrated Leakage Rate Test (ILRT) interval for no more than five (5) years. The exception is to allow ILRT testing within 15 years from the last ILRT, performed December 7, 1998.

NPPD requests Nuclear Regulatory Commission (NRC) approval of the proposed amendment by August 30, 2006. Approval by that date is needed to finalize the scope of the Cycle 23 refueling outage, currently scheduled to begin in October, 2006. The amendment will be implemented within 30 days of amendment issuance.

The proposed amendment is risk-informed. Therefore, NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis" has been followed, while using the methodology of Electric Power Research Institute (EPRI) Report "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," (EPRI TR-104285) and the 2001 Nuclear Energy Institute Interim Guidance.

This application represents a cost beneficial licensing change. The ILRT imposes significant expense on the station while the safety benefit of performing it within 10 years, versus 15 years is minimal. With the extension, the local leakage rate testing scope in both the Cycle 23 and Cycle 24 refueling outages would decrease.

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Attachment 1 provides a description of the proposed TS change, the basis for the amendment, the no significant hazards consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the environmental impact evaluation pursuant to 10 CFR 51.22. Attachment 2 provides the proposed changes to the current CNS TS on marked-up pages. Attachment 3 provides the revised TS pages in final typed format. The enclosure provides a detailed, plant specific risk assessment performed in support of this amendment request. No changes to Bases are involved with the proposed TS change.

The proposed TS changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 216 dated January 5, 2006, have been incorporated into this request. This request is submitted under oath pursuant to 10 CFR 50.30(b).

By copy of this letter, its attachments and enclosure, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

No new commitments are made in this submittal.

If you have any questions concerning this matter, please contact Paul Fleming, Licensing Manager at (402) 825-2774.

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

Executed on 1/30/06

Sincerely,

Randall K. Edington

Randall K. Edington
Vice President-Nuclear and
Chief Nuclear Officer

/rr

Attachments
Enclosure

cc: Regional Administrator w/ attachments and enclosure
US NRC – Region IV

Senior Project Manager w/ attachments and enclosure
US NRC – NRR Project Directorate IV-1

Senior Resident Inspector w/ attachments and enclosure
US NRC – CNS

Nebraska Health and Human Services w/ attachments and enclosure
Department of Regulation and Licensure

NPG Distribution w/out attachments and enclosure

CNS Records w/ attachments and enclosure

ATTACHMENT 1

License Amendment Request for a One-time Integrated Leakage Rate Test Interval Extension

Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

Revised Technical Specification Pages

5.0-16

1.0 Description

The proposed license amendment would revise Cooper Nuclear Station (CNS) Technical Specification (TS) section 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of no more than five years for the Type A, Integrated Leakage Rate Test (ILRT) interval. This revision is a one-time exception to the 10 year frequency of the performance-based leakage rate testing program for Type A tests as defined in Nuclear Energy Institute (NEI) document NEI 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," pursuant to 10 CFR 50, Appendix J, Option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT, last performed on December 7, 1998.

The technical analysis for the proposed license amendment is based on risk related and non-risk related considerations. A risk analysis performed by Nebraska Public Power District (NPPD) concluded that the increases in estimated person-rem and large early release frequency (LERF) are consistent with the guidance provided in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.174 and NUREG-1493. NPPD also demonstrated that defense-in-depth is provided by the small increase in the conditional containment failure probability, and by non-risk based considerations (for example, the ILRT and containment inspection history, and the ongoing Primary Containment Leakage Rate Testing (PCLRT) and Inservice Inspection (ISI)/IWE programs). The technical analysis provides the basis for the determination that the proposed amendment does not involve a significant hazards consideration as described in 10 CFR 50.92.

This application represents a cost beneficial licensing change. The ILRT imposes significant expense and hardship to the plant while the incremental safety benefit of performing the test within 10 years, versus 15 years, is minimal.

2.0 Proposed Change

CNS's TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," currently states:

A program shall establish 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. *Exemption from Appendix J to 10CFR Part 50 to allow reverse direction local leak rate testing of four containment isolation valves at Cooper Nuclear Station (TAC NO. M89769) (July 22, 1994).*

2. *Exemption from Appendix J to 10CFR Part 50 to allow MSIV testing at 29 psig and expansion bellows testing at 5 psig between the piles (Sept. 16, 1977).*

The proposed change would add a third exception to TS 5.5.12 to specify the date of the next required Type A test. Specifically, the added exception would state:

3. *Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", Section 9.2.3: The first Type A test performed after the December 7, 1998 Type A test shall be performed no later than December 7, 2013.*

3.0 Background

Description of CNS Containment System

CNS is a General Electric (GE), Model 4 Boiling Water Reactor (BWR-4) with a Mark I Primary Containment Pressure Suppression System. This system consists of the Drywell, which houses the reactor vessel and reactor coolant recirculation loops, the pressure Suppression Chamber which stores a large volume of water (known as the Suppression Pool), the connecting vent system between the Drywell and pressure Suppression Chamber, isolation valves, vacuum relief system, and containment cooling systems.

The Drywell is a low leakage steel pressure vessel designed to confine the reactor coolant that would be released during a postulated pipe rupture, and prevent the gross release of radioactive materials to the environment. It is designed for an internal pressure of 56.0 psig (62.0 psig maximum code allowable), 2.0 psid external pressure, and a maximum temperature of 281°F. The drywell is enclosed in reinforced concrete to provide radiological shielding during normal plant operations and resistance to deformation and buckling.

The Suppression Chamber is a steel pressure vessel, toroidal in shape, designed to hold a large volume of water for use as a heat sink for any postulated transient or accident conditions in which the normal heat sink is unavailable. The Suppression Chamber is located below, and completely encircling the Drywell. The Suppression Pool is the primary water source for the Emergency Core Cooling Systems of Core Spray and Residual Heat Removal Systems, and a secondary water source for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems. The water volume also serves as a heat sink for the turbine exhaust from the HPCI and RCIC Systems.

The Drywell and the Suppression Chamber were designed and constructed to the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The interior of the Drywell and the Suppression Chamber were coated with a coating which has been shown to satisfactorily withstand the temperatures and pressures of the steam environment postulated during a design-basis loss-of-coolant accident.

Testing Requirements of 10 CFR Part 50, Appendix J

The testing requirements of 10 CFR Part 50, Appendix J, provide assurance that leakage from the containment, including components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. The limitation of 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 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 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.

10 CFR Part 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." License Amendment No. 180 to the CNS Operating License, dated March 3, 2000, permitted the implementation of 10 CFR Part 50, Appendix J, Option B. The amendment added TS section 5.5.12, "Primary Containment Leakage Rate Testing Program," to require Type A, B and C testing in accordance with RG 1.163. RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and American Nuclear Standards Institute/American Nuclear Society Standard ANSI/ANS – 56.8-1994, subject to several regulatory positions in the guide. NEI 94-01 specifies an initial Type A test interval of 48 months, but allows an extended interval of ten years, based upon two consecutive successful tests.

The adoption of the Option B performance-based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Type A, B, and C tests but did not alter the basic method by which Appendix J leakage rate testing is performed. The 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 changes to the ILRT test frequency allowed by Option B do not result in an increase in containment leakage. Similarly, the proposed one-time extension of five years to the ILRT test frequency will not result in an increase in containment leakage.

The extended frequency interval for testing allowed by NEI 94-01 is based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program". The following statements regarding extending the test frequency are made in NUREG-1493, Section 10.1.2:

a. "Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one 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 and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements."

b. "Given the insensitivity of risk to containment leakage rate (Chapter 5) and the small fraction of leakage detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk."

c. "Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths."

d. "Based on the model of component failure with time, it has been found that performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. ... Since under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small."

10 CFR Part 50, Appendix J, Option B, Section V.B, "Implementation," allows exceptions to the guidelines of RG 1.163. That section states: "The regulatory guide or other implementation document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, the proposed TS change does not require an exemption from Option B.

4.0 Technical Analysis

<u>10 CFR Part 50, Appendix J, Option B Plant Specific Implementation</u>

As noted previously, License Amendment No. 180 permitted implementation of 10 CFR Part 50, Appendix J, Option B, by adding TS 5.5.12 to require Type A, B, and C testing in accordance with RG 1.163. RG1.163 endorses the methodology for complying with Option B identified in NEI 94-01. The surveillance frequency for Type A testing in NEI 94-01 is at least once every 10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart demonstrate the calculated performance leakage rate was less than 1.0 La) and consideration of the performance factors in NEI 94-01, Section 11.3.

The performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1. The performance leakage rate for Type A tests includes the Upper Confidence

Limit (UCL) plus the as-left minimum pathway leakage rate for all Type B and C pathways in service, isolated, or not lined up in their test position.

The two most recent Type A tests at CNS have been satisfactory with leakage rates for the December 1995 and December 1998 Type A tests being 0.30466 weight percent per day (wt% / day) and 0.28534 wt% / day, respectively. These results are less than the maximum allowable containment leakage rate (L_a, at P_a), of 0.635% containment air weight per day at a pressure of 58 psig. Based on these two consecutive successful tests, and License Amendment No. 180, the current ILRT interval requirement for CNS is 10 years.

Plant Testing and Inspection Programs

In addition to periodic Type A testing, various inspections and tests are routinely performed to assure primary containment integrity. These include Type B and C testing performed in accordance with Appendix J, Option B; inspection activities performed as part of the plant ISI/IWE Inservice Inspection program (includes inspection of drywell and suppression chamber surfaces and structural elements); and containment isolation valve inservice testing. The aggregate results of these tests and inspections provide a high degree of assurance of continued primary containment integrity.

Type B and Type C Program

The CNS Appendix J, Type B and Type C testing program is described in Engineering Procedure 3.40, "Primary Containment Leakage Rate Testing Program," and Surveillance Procedure 6.PC.501, "Primary Containment Local Leak Rate Tests." These procedures require testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the PCLRT Program as required by 10 CFR Part 50, Appendix J, Option B and RG 1.163.

The Type B and C test program provides a means to detect or measure leakage across pressure containing or leakage limiting barriers of the primary reactor containment. The results of the test program are used to ensure that proper maintenance and repairs are made on the primary reactor containment components over their service life. The Type B and C test program provides a means to protect the health and safety of plant personnel and the public by maintaining the leakage from these components below appropriate limits.

These components are tested with air or nitrogen at a pressure greater than or equal to 58 psig (P_a), except for the Main Steam Line Isolation Valves which are tested at 29 psig and the void between the bellows located in the main steam line and feedwater line penetrations which is tested at 5 psig.

As previously noted, Type B and Type C testing evaluate all but a small portion of potential containment leakage pathways. This amendment request does not affect the requirements of Type B or Type C tests scope, performance, or scheduling. Type B and

Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained.

Primary Containment Inspection Requirements / ISI / IWE Program

10 CFR Part 50, Appendix J, Option B, Section III.A states: "A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system." Since the interval for the Type A test has been previously extended to 10 years, this inspection is also conducted during two other refueling outages before the next Type A test in order to allow for early discovery of structural deterioration.

As noticed in the Federal Register dated August 8, 1996 (Vol. 61, No. 154; pages 41303-41312), the NRC revised 10 CFR 50.55a, "Codes and Standards," to incorporate by reference the 1992 Edition and Addenda of Subsections IWE and IWL of Section XI of the ASME Code. Subsections IWE and IWL specify the requirements for ISI of Class CC (concrete containments), and Class MC (metal containments) of light-water-cooled power plants. The amended rule became effective on September 9, 1996. The amended rule requires that licensees incorporate the new requirements into their ISI programs and to complete the first containment inspection within five years, i.e., no later than September 9, 2001. Any repair or replacement activity to be performed on containments after the effective date of September 9, 1996, has to be carried out in accordance with the respective requirements of Subsections IWE and IWL.

CNS completed the first IWE-required containment inspection in November 1998, and the second during the Cycle 21 refueling outage in spring 2003. The third inspection is scheduled to be conducted during the Cycle 24 refueling outage in spring 2008. The three inspection periods during the first inspection interval are:

First Period:	September 9, 1996 to September 8, 2001
Second Period:	September 9, 2001 to January 8, 2005
Third Period:	January 9, 2005 to May 8, 2008

The IWE containment inspection requirements are implemented at CNS through the "First Ten-Year Interval Containment Inspection Program for CNS." The general visual examination requirements specified in this containment inspection program satisfies the visual examination requirements specified in Option B.

IWE requirements include general visual inspection of the containment. These are conducted at CNS in accordance with Engineering Procedure 3.28.1.4, "General Visual Inspection of Containment Surfaces," or CNS-approved vendor procedures with similar requirements. A general visual inspection conducted by this procedure determines the structural integrity of the containment. The procedure includes inspections of the interior torus walls above the water level, the exterior of the torus above and below the water

level, vent pipes and downcomers above the water level, eight drywell penetration interior surfaces, eight Drywell-to-torus vent opening shield plates, interior and exterior of Drywell head, and verification that the vent header support pins are properly secured in place.

Engineering Procedure 3.28.1.4 identifies the following Acceptance Standards:

1. The following relevant conditions are unacceptable for continued service (Reference IWE-3510.1):
    a. Conditions that may effect the containment structural integrity or leak tightness.

2. Visual inspections that detect flaws or evidence of degradation that require evaluation may be supplemented by either surface or volumetric inspections, as necessary, to characterize the flaw.

3. Non-relevant conditions include fabrication marks, scratches, surface abrasion, material roughness, casting irregularities, and any other condition acceptable by the material, design, or manufacturing specifications for the component.

Indications or relevant conditions reported during visual inspection of ASME Code Class MC components are evaluated by a certified Level II or Level III Inspector. The evaluation, when possible, includes a review of any applicable pre-service and inservice inspection records, and fabrication records. The evaluations are reviewed by the Inservice Inspection Engineer and any conditions exceeding the allowable standards of ASME Section XI are entered into the Corrective Action Program.

Containment inspections will continue to be performed during the proposed 5-year extension of the interval (December 2008 through December 2013) in accordance with the CNS PCLRT and ISI/IWE program as applicable.

Approved Alternatives to Subsection IWE Requirements

The following two relief requests, associated with the PCLRT Program, were approved by the NRC for application at CNS.

1. Relief Request No. RC-02

    IWE-2500, Table IWE-2500-1 requires seals and gaskets on airlocks, hatches, and other devices to be visually examined (VT-3) once each interval to assure containment leak-tight integrity.

    As an alternative, the leak tightness of seals and gaskets will be tested in accordance with 10 CFR 50, Appendix J, Type B testing. No additional alternatives to the visual examination, VT-3, of the seals and gaskets will be performed.

2. Relief Request No. RC-06, Revision 1

   IWE-5240 invokes the requirements of IWA-5240 as applicable following repair, replacement, or modification. IWA-5240 requires a VT-2 visual examination in conjunction with the pressure test.

   Table IWE-2500-1, Category E-P, identifies the examination method as 10 CFR Part 50 Appendix J, and does not specifically require a VT-2 visual examination. 10 CFR Part 50 Appendix J, provides the requirements for testing, as well as the acceptable leakage criteria. These tests are performed by qualified test personnel using calibrated equipment to determine the leakage rate. In addition, 10 CFR 50.55a(b)(2)(x)(E) requires a general visual examination of the containment each inspection period. This inspection would identify any structural degradation that may contribute to leakage. A VT-2 visual examination will not provide additional assurance of safety beyond that of current Appendix J requirements.

The request for the one-time extension of the ILRT interval has no affect on the performance of the required alternate testing activities described in these relief requests.

Inservice Testing Program

Containment isolation valves (CIVs) are tested, as applicable, in accordance with the CNS Third Interval Inservice Testing Program, as required by 10 CFR 50.55a. This testing ensures operational readiness of the CIVs, such that they will perform their containment isolation function when called upon.

Painting and Coatings Program

Surface preparation, coating application, and coating inspection requirements for Drywell and Suppression Chamber surfaces are controlled by the CNS Painting and Coatings Program. CNS Procedures 3.41, "Painting and Coatings Program," and 7.0.15.1, "Service Level I Coating," provide the necessary controls for maintenance of safety-related coatings applied to the Drywell and Suppression Chamber surfaces to ensure the coating protects these surfaces from corrosion, erosion, and mechanical damage or wear.

Plant Operational Performance

During power operation the CNS primary containment is inerted with nitrogen to maintain oxygen concentration within TS limits. As a result, the primary containment is maintained at a slightly positive pressure during power operation. Drywell pressure is continuously recorded and is verified to be within limits by TS surveillance every 12 hours from the control room. Maintaining the containment pressurized at power, and frequently monitoring the pressure, assures that gross containment leakage that may develop during power operation will be detected.

NRC Information Notice 92-20

NRC Information Notice (IN) 92-20, "Inadequate Local Leak Rate Testing," 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.

The main steam and feedwater testable penetrations at CNS consist of a double layered metal bellows. The inboard high pressure side of the bellows is subject to Drywell pressure. Therefore, the bellows is tested in its entirety during the performance of the Type A test. The bellows are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage, and possible ruptures of the bellows.

Numerous tests related to IN 92-20 have been performed at CNS since startup. The following is a summary of those tests.

Plant startup to 1992:	Eight sets of LLRTs conducted between the plies and five ILRTs conducted show the bellows to be acceptably leak-tight.
1993:	One set of LLRTs between the plies and a "pass/fail" helium test applied from the direction of pressure in containment.
1994 to 1997:	Two sets of LLRTs between the plies and an ILRT in 1995.
1998 to present:	Performance of an ILRT in 1998 and performance of LLRTs between the plies in 1998, 2000, and 2005.

Successful LLRTs of the bellows, successful ILRTs, and an acceptable helium test demonstrates adequate testing and leak-tight integrity of the bellows at CNS.

PSA-ES067, Risk Impact Assessment of Extending Containment Type A Test Interval for Cooper Nuclear Power Station, provided as Enclosure to this submittal, considers the potential failure of containment bellows assemblies.

Through-wall Torus Shell Crack at James A. Fitzpatrick Nuclear Power Plant

A through-wall torus shell crack was discovered at the James A. Fitzpatrick Nuclear Power Plant (JAF) on June 27, 2005. NPPD reviewed the issue for applicability to CNS, and documented the results in Corrective Action Program Condition Report CR-CNS-2005-06052. The review concluded that the condition at JAF is not applicable to CNS due to differences in the internal and external torus structures between CNS and JAF. Those differences include:

1. Spargers are not installed on the HPCI and RCIC exhaust lines in the JAF Torus. Spargers are installed on these lines in the CNS Torus.

2. The supports for the exhaust lines at CNS are more robust than those at JAF.

NRC Information Notice 88-82

NRC IN 88-82 documented degradation of torus coatings. Supplement 1 to IN 88-82 recommended underwater inspection and repair. NPPD began conducting inspections in response to IN 88-82 in 1989. In 1996 the inspections were incorporated into the Inservice Inspection (ISI/IWE) Program. IWE requires the performance of VT-3 visual examination of the interior submerged surfaces of the torus. The required examinations and their schedule are contained in the First Ten-Year Interval Containment Inspection Program.

IWE requirements include an inspection of the submerged portions of the torus. This is conducted at CNS by Engineering Procedure 3.28.1.5, "Visual Inspection of Containment Surfaces, VT-1/3," or CNS-approved vendor procedures with similar requirements. A visual inspection is conducted in accordance with this procedure to determine the condition of the part, component, or surface examined. The inspections are looking for conditions such as cracks, wear, corrosion, erosion, or physical damage on the surface of the part or component.

The following are identified as Acceptance Standards in Engineering Procedure 3.28.1.5:

1. The following VT-1 and VT-3 relevant conditions are unacceptable for continued service (Reference IWE-3510.2, -3510.3, -3512):

    a. Evidence of flaking, blistering, peeling, discoloration, and other signs of distress in areas that are painted or coated.

    b. Evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of distress in areas that are not painted or coated.

2. Visual examinations that detect flaws or evidence of degradation that require evaluation may be supplemented by either surface or volumetric examinations, as necessary, to characterize the flaw.

3. Non-relevant conditions include fabrication marks, scratches, surface abrasion, material roughness, casting irregularities, and any other conditions acceptable by the material, design, or manufacturing specifications for the component.

Indications or relevant conditions reported during visual inspections of ASME Code Class MC components are evaluated by a certified Level II or Level III Inspector. The

evaluation, when possible, includes a review of any applicable preservice and inservice inspection records, and fabrication records. The evaluations are reviewed by the Inservice Inspection Engineer and any condition exceeding the allowable standards of ASME Section XI are entered into the Corrective Action Program.

The coatings on the submerged surfaces of the torus at CNS have degraded since original construction. The discrete areas of degraded coating have been repaired by the application of an underwater epoxy coating as necessary to meet the requirements and recommendations of IN 88-82 and Supplement 1 to IN 88-82.

NPPD will continue to inspect the torus interior surfaces, including the submerged surfaces of the pressure boundary, and repair areas of degraded coating to ensure that these surfaces are adequately maintained throughout the life of the plant. These inspections are part of the Inservice Inspection (ISI/IWE) Program "Successive Examinations." The first of these inspections was performed during the Cycle 22 refueling outage, conducted during January and February, 2005. The next required IWE Code "Successive Examination" is required during the Cycle 24 refueling outage, currently scheduled for spring 2008. CNS is continuing to monitor the industry and evaluate options for torus recoat in the future.

Plant Specific Risk Assessment

An evaluation was performed to assess the risk impact of a one-time extension of the containment ILRT frequency from 10 years to 15 years for CNS. The risk assessment was performed using the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, and the NRC regulatory guidance of RG 1.174 on the use of Probabilistic Risk Assessment findings and risk insights in support of a request to change the licensing basis of a plant. In addition, a risk assessment was performed using two more recent studies for comparison purposes. The methodologies used are those presented in the NEI Interim Guidance, and in EPRI TR-1009325. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent studies.

The findings of the CNS assessment confirm the general findings of previous studies on a plant specific basis, including severe accident category frequencies, the containment failure modes, the TS allowed leakage, and the local population surrounding CNS. The following are the conclusions regarding the risk impact of extending the Type A ILRT test from 10 to 15 years:

1. There is no change in the at-power Core Damage Frequency (CDF) associated with the ILRT test interval extension. Therefore, this is within the RG1.174 acceptance guidelines.

2. RG 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 those that result

in increases of CDF less than 1E-6/yr and increases in Large Early Release Frequency (LERF) less than 1E-7/yr. (LERF is the relevant criterion since CDF is not impacted by the ILRT). The increase in LERF resulting from a change in the ILRT test frequency from one in 10 years to one in 15 years was determined to be between 2.41E-10/yr and 2.63E-9/yr. The range in LERF is based on the use of three different methodologies in the risk analysis. Therefore, increasing the ILRT interval from 10 years to 15 years results in a very small change to the CNS risk profile.

3. The change in ILRT test frequency from one in 10 years to one in 15 years increases the total integrated plant risk, in terms of person-rem/yr within 50 miles, by between 0.0009% and 0.007%. The range in total integrated plant risk is based on the use of three different methodologies in the risk analysis. The increase in risk was determined to be of a magnitude that NUREG-1493 indicates is imperceptible. Therefore, the change in risk impact, when compared to that of other severe accident risks, is negligible.

4. The change in Conditional Containment Failure Probability was less than 1%. This is insignificant and reflects sufficient defense-in-depth.

5. Incorporating external event impacts into this analysis does not change the conclusion of this risk assessment that increasing the ILRT interval from 10 years to 15 years is an acceptable plant change from a risk perspective.

The details of the risk assessment are contained in the Enclosure to this submittal.

Discussion of Precedents

NPPD considers Amendment No. 227 to the operating license of Vermont Yankee Nuclear Power Station (VYNPS), and Amendment No. 187 to the operating license for the Edwin I. Hatch Nuclear Plant (HNP) as valid precedents. Like CNS, both of these plants are Model 4 Boiling Water Reactors with Mark I pressure suppression systems, consisting of the Drywell, the pressure Suppression Chamber, and the interconnecting piping.

Also, the requirements for containment leakage rate testing and inspection at CNS are similar to those at VYNPS and HNP. CNS, VYNPS, and HNP perform Type A, Type B, and Type C testing in accordance with 10 CFR 50, Appendix J, Option B, and perform inspections in accordance with Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code.

Like VYNPS and HNP, CNS operates with a positive pressure in containment, which is monitored, and provides an indication of leakage if a pressure decrease occurs.

The risk assessments performed for VYNPS, HNP, and CNS used the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, and the regulatory guidance from NRC Regulatory Guide 1.174.

Conclusion

Based on the previous ILRT tests conducted at CNS which confirm that the reactor containment structure exhibits extremely low leakage, NPPD concludes that the 5-year increase in the interval represents minimal risk to increased leakage. The risk is minimized by continued Type B and C testing performed in accordance with Option B of Appendix J, inspection activities performed as part of the plant Inservice Inspection (ISI/IWE) program, quality control of Drywell and Suppression Chamber surface coating repair/replacement, containment isolation valve inservice testing, and by operating experience with a containment that normally operates at a positive pressure (i.e., the pressure from containment inerting). In the aggregate these provide continuing confidence in containment integrity.

This experience is supplemented by studies, including the CNS risk analysis (Enclosure). That analysis has determined that the increase in risk from the requested, one-time, five-year extension is very small.

NPPD therefore concludes that the requested five-year extension of the ILRT interval does not endanger public health and safety.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of no significant hazards posed by issuance of the amendment. Nebraska Public Power District (NPPD) has evaluated this proposed amendment with respect to the criteria in 10 CFR 50.92(c). The following is the evaluation required by 10 CFR 50.91(a)(1).

NPPD is requesting an amendment of the operating license for the Cooper Nuclear Station (CNS). The requested amendment is to allow a one-time extension of the frequency for performance of the next primary containment integrated leak rate test (ILRT). 10 CFR Part 50, Appendix J, requires the performance of an ILRT. The current test interval of 10 years, based on past performance, is proposed to be extended, on a one-time basis, to 15 years from the last ILRT test. The proposed extension is reflected by a revision of Technical Specification Section 5.5.12, "Primary Containment Leakage Rate Testing Program."

The following reflects the evaluation of the changes against the three criteria of 10 CFR 50.92(c). The evaluation supports a finding of "no significant hazards" for the proposed amendment.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

This license amendment proposes to revise the Technical Specifications to allow for a one-time extension of the ILRT interval from 10 years to 15 years. The containment function is solely to mitigate the consequences of an accident. No design basis accident is initiated by a failure of the containment leakage mitigation function. The extension of the ILRT will not create any adverse interactions with other systems that could result in initiation of a design basis accident. Continued containment integrity is also assured by the established programs for local leakage rate testing and inservice inspections which are unaffected by the proposed change. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 to 15 years. The increase in risk in terms of person-rem per year within 50 miles resulting from accidents was determined to be of a magnitude that NUREG-1493 indicates is imperceptible. NPPD has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed extension was determined to be within the guidelines published in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NPPD has determined that the increase in conditional containment failure probability from reducing the ILRT frequency from one test in 10 years to one test in 15 years would be insignificant.

Therefore, the probability of occurrence or the consequences of an accident previously analyzed are not significantly increased.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No

The proposed extension of the current interval for the ILRT does not involve any change to the design or operation of any plant structure, system, or component (SSC). The plant will continue to be operated in the same manner. Since no changes to the design or operation of the plant are

being made, the proposed one-time extension of the ILRT does not result in a new failure mode for an accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

Response: No.

The proposed extension to the ILRT test interval will not result in a change to the design or operation of any plant SSC used to shutdown the plant, initiate Emergency Core Cooling Systems, or isolate the primary or secondary containment. Thus, the change will not impact the ability of CNS to mitigate any accident or transient. NUREG-1493, a generic study of the effects of extending containment leakage testing, documented that an extension in the ILRT interval from three per 10 years to one per 20 years resulted in an imperceptible increase in risk to the public. NUREG-1493 generically concluded that the design containment leakage rate contributes about 0.1 percent to the individual risk, and that the decrease in the ILRT frequency would have a minimal effect on this risk since 95% of the potential leakage paths are detected by Type B and Type C testing. A risk assessment using the current CNS Probabilistic Safety Assessment internal events model concluded that the risk associated with this change is very small and not risk significant.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety

Based on the above, NPPD concludes that the proposed amendment presents no 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.

5.2 Applicable Regulatory Requirements/Criteria

1. 10 CFR Part 50, Appendix J, Option B. This regulation requires that licensees' primary reactor containments meet the leakage rate requirements as delineated by Appendix J. This requirement is met by performance of Type A, B, and C leakage rate testing on the primary containment and its associated components (e.g., valves, penetrations). The leakage rate test results are compared to allowable leakage rate acceptance criteria set forth by Appendix J. CNS TS 5.5.12, "Primary Containment Leakage Rate Testing Program," invokes Appendix J requirements.

TS 5.5.12 requires that the leakage rate testing of the containment be performed in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by NRC-approved exemptions, and in accordance with NRC RG 1.163. Regulatory Position C.1 of RG 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01. Section 11.0 of NEI 94-01 references Section 9.0, which would require that ILRTs be performed for CNS within 10 years plus 15 months from the date of their last performance.

CNS will continue to comply with the requirements of 10 CFR 50, Appendix J, with the proposed ILRT extension. No other regulations or TS are affected by the proposed amendment.

2. Updated Safety Analysis Report (USAR) Section V, "Containment." Section V of the CNS USAR provides licensing basis information for the reactor containment vessel. Subsection V-2.5 describes pre-operational and subsequent leakage rate testing of the containment. This subsection states that subsequent local (Type B and C) leakage rate tests are in accordance with the requirements of 10 CFR 50, Appendix J and Technical Specifications.

   The validity of this statement is unaffected by the proposed amendment since the proposed extension will only apply to the current ILRT interval and will not alter the accuracy of the statements as descriptions of normal requirements. Additionally, the proposed amendment does not affect any other aspect of the ILRT, such as test methodology, pressure, or acceptance criteria. As a result CNS will continue to comply with the requirements of USAR Section V with the proposed ILRT extension.

3. USAR Section XIV, "Safety Analysis." Section XIV of the CNS USAR provides descriptions of the licensing basis accident analyses for CNS including the relevant parameters for the analyses.

   The proposed amendment only involves the ILRT interval and does not affect any parameters, such as pressure or leakage rate, that can affect the results of these analyses. As a result, CNS will continue to comply with the requirements of USAR Section XIV with the proposed ILRT extension.

4. USAR Appendix F, Conformance to AEC Proposed General Design Criteria, Criterion 54, "Containment Leakage Rate Testing," states:

   "Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance."

The proposed extension does not alter the design of the containment. As a result, the ability to conduct leakage rate testing at design pressure would not be adversely impacted. Thus, the requirements of this criterion will continue to be met with the proposed ILRT frequency extension.

5. USAR Appendix F, Conformance to AEC Proposed General Design Criteria, Criterion 55, "Containment Periodic Leakage Rate Testing," states:

   "The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime."

   The proposed extension does not alter the design of the containment. As a result, the ability to perform periodic testing of the containment would not be adversely impacted. Thus, the requirements of this criterion will continue to be met with the proposed ILRT frequency extension.

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 detrimental to the common defense and security or to the health and safety of the public.

6.0 Environmental Consideration

A review has determined that the proposed amendment would change a 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 needs to be prepared in connection with the proposed amendment.

7.0 References

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

2) Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

3) An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, July 1998.

4) Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.

5) J. Haugh, John Gisclon, W. Parkinson, Ken Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Intervals", Rev. 4, EPRI, November 2001.

6) Amendment No. 180 to the CNS Operating License, dated March 3, 2000 (TAC NO. MA6877).

7) PSA-ES067, Risk Impact Assessment of Extending Containment Type A Test Interval for Cooper Nuclear Power Station.

8) Regulatory Guide 1.163, Performance Based Containment Leak-Test Program, dated September 1995.

9) ANSI/ANS 56.8-1994, Containment system Leakage Testing Requirements.

10) Engineering Procedure 3.40, Primary Containment Leakage Rate Testing Program.

11) Surveillance Procedure 6.PC.501, Primary Containment Local Leak Rate Tests.

12) NRC Information Notice 88-82, Torus Shells with Corrosion and Degraded Coatings in BWR Containments.

13) Precedent: Vermont Yankee Nuclear Power Station, Facility Operating License DPR-28, Amendment No. 227, dated August 31, 2005 (TAC No. MC4662).

14) Precedent: Edwin I. Hatch Nuclear Plant, Unit 2, Facility Operating License number NPF-5, Amendment Number 187, dated February 1, 2005 (TAC Nos. MC2761).

15) First Ten-Year Interval Containment Inspection Program for Cooper Nuclear Station.

16) Cooper Nuclear Station Third Interval Inservice Testing (IST) Program.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION REVISIONS (MARK-UP)

COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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 system(s) supported by the inoperable support system is also inoperable; or

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

3. A required system redundant to support system(s) for the supported systems b.1 and b.2 above is also inoperable.

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

5.5.12 Primary Containment Leakage Rate Testing Program

a. A program shall establish 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. Exemption from Appendix J to 10CFR Part 50 to allow reverse direction local leak rate testing of four containment isolation valves at Cooper Nuclear Station (TAC NO. M89769) (July 22, 1994).

   2. Exemption from Appendix J to 10CFR Part 50 to allow MSIV testing at 29 psig and expansion bellows testing at 5 psig between the piles (Sept. 16, 1977).

   3. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 9.2.3: The first Type A test performed after the December 7, 1998 Type A test shall be performed no later than December 7, 2013.

insert

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58.0 psig. The containment design pressure is 56.0 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.635% of containment air weight per day.

(continued)

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION REVISIONS
(FINAL TYPED)

COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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 system(s) supported by the inoperable support system is also inoperable; or

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

3. A required system redundant to support system(s) for the supported systems b.1 and b.2 above is also inoperable.

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

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3. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 9.2.3: The first Type A test performed after the December 7, 1998 Type A test shall be performed no later than December 7, 2013.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58.0 psig. The containment design pressure is 56.0 psig.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.635% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq$ 12 scfh when tested at $\geq$ P_a.

b. Overall air lock leakage rate is $\leq$ 0.23 scfh when tested at $\geq$ 3.0 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Correspondence Number: NLS2006002

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		

ENCLOSURE

RISK IMPACT ASSESSMENT OF EXTENDING CONTAINMENT TYPE A TEST INTERVAL FOR COOPER NUCLEAR POWER STATION

COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

PROBABILISTIC SAFETY ASSESSMENT

COOPER NUCLEAR STATION

ENGINEERING STUDY

Title: Risk Impact Assessment of Extending Containment Type A Test Interval for Cooper Nuclear Power Station

Log No.: PSA-ES067

	Signature	Date
Revision 1 Prepared By: Reliability Engineer	Glen Seeman	12/14/2005
Reviewed By: Reliability Engineer	Kent Sutton	12-14-05
Approved By: Supervisor, Reliability Engineering	Kent Sutton	12-14-05

Revisions:

Number	Description / Prepared By / Date	Reviewed By	Date	Approved By	Date
0	Original Issue Rick Anoba/Glen Seeman/12-8-2005	Kent Sutton	12-8-05	Kent Sutton	12-8-05
1	Correct typographical errors/ As above	As Above		As Above	

NOTE:
Signatures by the Preparer and Reviewer indicate that the information provided in this Engineering Study has been validated to be correct.

Table of Contents

List of Tables and Figures

EXECUTIVE SUMMARY

An evaluation was performed to assess the risk impact of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for the Cooper Nuclear Station (CNS). The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for CNS. The proposed change would impact testing associated with the current surveillance test for Type A leakage. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3]. In addition, for comparison purposes, the risk assessment was also performed using two more recent (although not yet issued in final, approved form) studies. These methodologies are presented in the NEI Interim Guidance [23], and in EPRI TR-1009325 [30]. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent methodologies.

The findings for the CNS assessment confirm the general findings of previous studies on a plant specific basis, including severe accident category frequencies, the containment failure modes, the Technical Specification allowed leakage, and the local population surrounding the CNS station. The following conclusions are provided with regard to the risk impact of extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

- Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is between 2.41E-10/yr and 2.63E-9/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1E-7/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the CNS risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk (total dose to the public) by between 0.0009% and 0.007%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- This change in Conditional Containment Failure Probability (CCFP) of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

- Incorporating external event impacts into this analysis does not change the conclusion of this risk assessment (i.e., increasing the CNS ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

Risk Impact Assessment of Extending Containment Type A Test Interval for Cooper Nuclear Power Station

PSA-ES067, Rev. 0

1. Introduction

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for the Cooper Nuclear Station (CNS). The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for CNS. The proposed change would impact testing associated with the current surveillance test for Type A leakage. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3]. In addition, for comparison purposes, the risk assessment was also performed using two more recent (although not yet issued in final, approved form) studies. These methodologies are presented in the NEI Interim Guidance [23], and in EPRI TR-1009325 [30]. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent studies. Therefore, the calculations and results from these analyses are presented in Sections 9 and 10, with previous EPRI TR-104285 results referenced as necessary for efficient reporting of the study results.

2. Limitations

This PRA Engineering Study evaluates the risk significance of extending ILRT test frequency based on Internal Events PRA Model Version 1996b [29] and the 1998 update of the Level II Analysis [24]. External events were addressed as a sensitivity study but not included in the final results. Since CNS does not have a Level III analysis, the population doses specified in NUREG/CR-4551 were used for this analysis. The NUREG/CR-4551 doses were adjusted for the population surrounding the CNS and the CNS ILRT leakage criteria.

3. Background

10CFR50, Appendix J, Option B, allows individual plants to extend the Type A Integrated Leak Rate Test (ILRT) surveillance test interval from three-in-ten years to at least once per ten years. The revised Type A test frequency is based on an acceptable performance history defined as two

consecutive periodic Type A tests at least 24 months apart in which the calculated leakage performance was less than 1.0La. CNS meets these requirements.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rule making to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rule making basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [2].

NUREG-1493 [4] analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing and determined that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day results in a small increase in total population exposure. In addition, increasing the leak rate to 50 percent per day increases the total population risk by less than 1 percent. Consequently, extending the ILRT interval should not result in a substantial increase in risk. The current analysis is being performed to confirm these conclusions based on CNS specific models and available data.

EPRI TR-104285 (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals) is a follow-on report to NUREG-1493 that provides a methodology for use in preparing PRA analysis to support a submittal to extend ILRT test intervals. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct a general visual inspection of the accessible areas of the interior of the containment in accordance with Subsection IWE once each period. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B and Type C local leak tests performed to verify the leak-tight integrity of containment penetration valves, air locks, seals, and gaskets are also not affected by the change to the Type A test frequency.

4. Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time 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. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated. In addition, the total risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter.

5. 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 EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current CNS PRA model that includes the results from the CNS Level II analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release).

The four general steps of this risk assessment are as follows:

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

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 LERF in accordance with Regulatory Guide (RG) 1.174 [3] and compare with the acceptance guidelines of RG 1.174.

This approach is based on the information and methodology contained in the previously mentioned studies and is consistent with the following:

- Other industry risk assessments for ILRT test interval extensions. The CNS assessment uses population dose as one of the risk measures. The other risk measures used in the CNS assessment are LERF, and Conditional Containment Failure

Probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.

- EPRI TR-104285 and NUREG-1493. The CNS assessment uses information from NUREG-1273 [6] regarding the low percentage of containment leakage events that would only be detected by an ILRT as input to calculate the increase in the pre-existing containment leakage probability due to the testing interval extension.

- The approach used in the Indian Point 3 risk-informed submittal for a one-time extension of the Type A test interval. The CNS evaluation uses similar ground rules and methods to calculate changes in risk metrics [14]. NRC granted approval to Indian Point 3 by License Amendment No. 206 dated April 17, 2001 (TAC No. MB0178) [21].

6. Inputs

This section summarizes the general resources available as input (Section 6.1) and the plant specific resources required (Section 6.2).

6.1 General Resources Available

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

1) NUREG/CR-3539 [7]
2) NUREG/CR-4220 [8]
3) NUREG-1273 [6]
4) NUREG/CR-4330 [9]
5) EPRI TR-105189 [10]
6) NUREG-1493 [4]
7) EPRI TR-104285 [2]

1. NUREG/CR-3539 [7]

The study is applicable because it provides one basis for the threshold that could be used in the Level II PRA for the size of containment leakage that is considered significant and to be included in the model. 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 as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on Light Water Reactor (LWR) accident risks is relatively small.

2. NUREG/CR-4220 [8]

This 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. NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand License Event Reports (LERs), ILRT reports, and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. It is the latter category that is applicable to containment isolation modeling and that is the focus of this risk assessment.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events.

3. NUREG-1273 [6]

The study is applicable because it is a subsequent study to NUREG/CR-4220 which undertook a more extensive evaluation of the same 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.

4. NUREG/CR4330 [9]

This study provides an assessment of the impact of different containment leakage rates on plant risk. 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. EPRI TR-105189 [10]

This study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation

(using the EPRI ORAM software) of the impact of extending ILRT and LLRT test intervals on shutdown risk for two reference plants (a BWR-4 and a PWR).

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. Extending the ILRT frequency from 3 per 10 years resulted in a reduction in shutdown CDF of approximately 1E-7/yr. This risk reduction is due to the following issues:

- Reduced potential of vessel drain down events
- Reduced time in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS-draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in safety due to extending the ILRT test interval.

6. NUREG-1493 [4]

The 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. 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 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2-1.0%) population risk.

NUREG-1493 used information from NUREG-1273 regarding the low percentage of containment leakage events that would only be detected by an ILRT in the calculation of the increase in the pre-existing containment leakage probability due to the testing interval extension. NUREG-1493 makes the following assumptions in this probability calculation:

- The average time that a pre-existing leakage may go undetected increases with the length of the testing interval (and is ½ the length of the test interval).
- Only 3% of all pre-existing leaks can be detected only by an ILRT (i.e., and not by LLRTs).

This same approach that was used in a previously approved ILRT test interval extension submittal [14, 21] is also proposed here for the CNS ILRT test interval extension risk assessment.

7. EPRI TR-104285 [2]

The study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. 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 II models with NUREG/CR-4551 Level III 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 used a simplified Containment Event Tree to subdivide representative core damage frequencies into eight (8) 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. Containment isolation failures not identified by LLRT (e.g., isolation failures due to testing or maintenance)
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "These study results show that 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 0.02 person-rem per year..."

6.2 Plant Specific Inputs

The information used to perform the CNS ILRT Extension Risk Assessment includes the following:

- Population Dose Calculations by release category.
- CNS PRA Model
- ILRT results to demonstrate adequacy of the administrative and hardware issues. The two most recent Type A ILRT tests for CNS were successful, so the current Type A test interval is 10 years.

Population Dose Calculations

The population dose is calculated by using Peach Bottom data provided in NUREG/CR-4551 [5] and adjusting the results for CNS. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551[5].

The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 [5] and are reproduced in Table 6-1 for references purposes. Table 6-2 summarizes the calculated population dose associated with each APB from NUREG/CR-4551[5].

Population Estimate Methodology

The person-rem results in Table 6-2 can be used as an approximation of the dose for CNS if it is corrected for the population surrounding CNS and the difference in Technical Specifications leak rate. For the updated population estimate, data is available for population by county from the US Census Bureau on the website (http://quickfacts.census.gov/qfd/states/27000.html). This data was used to estimate the population within a 50-mile radius of the plant. If any part of a county fell within the 50-mile radius (based on a review of a map containing a mileage scale and state/county borders), then the entire population was included in the population estimate. For example, the population of Lancaster County is 250,291 people. However 225,000 people of this county live in the city of Lincoln which is outside the 50-mile radius of the plant. The results of this population estimate are presented in Table 6-3.

The total population shown in Table 6-3 is compared to the total population that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that was applied to the APBs to get dose estimates for CNS.

Total CNS Population = 4.87E+5 [Table 6-3]
Peach Bottom Population from NUREG/CR-4551 = 3.02E+6 [5]
Population Dose Factor = 4.87E+5 / 3.02E+6 = 0.161

This population dose factor was applied to the APBs from NUREG/CR-4551. Additionally, a second correction factor is also required to be applied to the NUREG/CR-4551 calculation to account for differences in the Technical Specification [*A5*] leakage value for Accident Progression Bin 8. The Technical Specification containment available leak rate for CNS is 0.635% (L_a^M) versus the 0.5% (L_a^{PB}) for the NUREG/CR-4551 plant, Peach Bottom. Therefore, the leakage (L_a^{PB}) person-rem calculated for Peach Bottom that is scaled by population for the CNS analysis must be multiplied by a factor of 1.27 (L_a^{CNS} / L_a^{PB}) to account for the differences in Technical Specification leakage rates. Table 6-4 shows the results of applying the population dose factor and the allowable leakage factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for CNS. Since CNS has a lower power rating than Peach Bottom, it is reasonable to assume that the dose release for the CNS containment would be lower. The dose corrections conservatively neglect the reduction factor for plant power rating.

CNS PRA Model

The version of the PRA model used for the CNS ILRT Extension Risk Assessment is comprised of a Level I PRA Model, CNS PRA 1996b, and Level II PRA model developed in 1998. This version of the PRA model addresses accidents initiated by internal events at full power, and containment responses to these accidents. In Reference [28], the NRC concluded that the use of this version of the model was appropriate for Risk-Informed In-service Inspection (RI-ISI) Assessment. This conclusion was based on inputs from References [26] and [27]. Reference [26] indicated that, although the model was undergoing modifications to address peer review/certification comments, the recommended improvements would not have a significant impact on the CNS RI-ISI Assessment. Specifically, although not incorporated into the submitted version of PRA model, the recommended improvements in the Initiating Events Analysis, the Data Analysis, and Human Reliability Analysis would not have a significant impact on the CNS RI-ISI Assessment.

For the CNS ILRT Extension Risk Assessment, the controlling risk parameters are the frequency of the intact containment release category relative to the total core damage frequency, and the magnitude of the intact containment population dose relative to the total dose (i.e., the higher the intact frequency relative to the total core damage frequency and/or the intact population dose relative to the total dose, the larger the impact on ILRT risk). Any increases in initiating event frequencies, basic event probabilities, or human failure event probabilities would have a tendency to reduce the relative frequency of the intact containment release category relative to the total core damage frequency. Any reductions in initiating event frequencies, basic event probabilities, or human failure event probabilities would have a tendency to increase the relative frequency of the intact containment release category relative to the total core damage frequency. However, since the frequency of the intact containment category for CNS is about four percent of the total core damage frequency, it is unlikely that any reasonable reductions in initiating event frequencies, basic event probabilities, or human failure event probabilities would have any significant impact on ILRT risk. It is therefore concluded that this version of the CNS PRA model is adequate for ILRT Extension Risk Assessment.

As shown in Table 6-5, the 1998 update of the CNS Level II PRA was used to quantify percent contributions for radiological release categories based on containment failure mode. The CNS PRA 1996b model was used to quantify frequencies for the radiological release categories for ILRT Extension. For the CNS PRA 1996b model, the core damage frequency (CDF) is 1.3E-5 per-yr and the Large Early Release Frequency (LERF) is 5.6E-7 per-yr. For the ILRT Extension, the frequencies for the radiological release categories in Table 6-6 were quantified by multiplying the CNS PRA 1996b CDF by the percent contributions on Table 6-5.

EPRI Release Category Definition

Table 6-7 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 6-8 defines the mapping of Peach Bottom Accident Progression Bins (APB) to EPRI accident class. As defined on Table 6-1, for APB #3, core damage occurs followed by vessel breach. The containment (drywell) fails early in the event (i.e., either before core damage, during

core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating is possible). Therefore, the release associated with APB #3 is assumed to be equivalent to the release associated with EPRI Classes 2, 7, and 8 from Table 6-7.

Table 6-9 defines the mapping of CNS Containment Failure Modes to EPRI release class. The CNS "Containment Isolation Failure" bin may be due to containment isolation or containment bypass. Therefore, the release associated with this CNS bin is assumed to be equivalent to the release associated with EPRI Classes 2 and 8 from Table 6-7.

Table 6-1 COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTION [5]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

Collapsed APB Number	Description
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Table 6-2 CALCULATION OF PEACH BOTTOM (NUREG/CR-4551) POPULATION DOSE

Collapsed Accident Progression (APB) Number	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 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	0.021	0.1659	9.55E-8	1.74E+6
2	0.0066	0.05214	4.77E-8	1.09E+6
3	0.556	4.3924	1.48E-6	2.97E+6
4	0.226	1.7854	7.94E-7	2.25E+6
5	0.0022	0.01738	1.30E-8	1.34E+6
6	0.059	0.4661	2.04E-7	2.28E+6
7	0.118	0.9322	4.77E-7	1.95E+6
8	0.0005	0.00395	7.99E-7	4.94E+3
9	0.01	0.079	3.86E-7	2.05E+5
10	0	0	4.34E-8	0
Totals	1.0	7.9	4.34E-6	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided 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 2.5-6. 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.

Table 6-3 POPULATION WITHIN 50 MILES OF CNS

State	County	Population (2000 Census)
Iowa	Mills	14,547
	Montgomery	11,771
	Page	16,976
	Taylor	6,958
	Fremont	8,010
Kansas	Brown	10,724
	Doniphan	8,249
	Marshall	10,965
	Nemaha	10,717
Missouri	Andrew	16,492
	Atchison	6,430
	Holt	5,351
	Nodaway	21,912
Nebraska	Cass	24,334
	Gage	22,993
	Johnson	4,488
	Lancaster	250,291
	Richardson	9,531
	Nemaha	7,576
	Otoe	15,346
	Pawnee	3,087
Total		486,748

Table 6-4 CALCULATION OF CNS POPULATION DOSE AT 50 MILES

Peach Bottom Accident Progression Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain Cooper Population Dose	CNS Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+6	0.161	2.80E+5
2	1.09E+6	0.161	1.76E+5
3	2.97E+6	0.161	4.78E+5
4	2.25E+6	0.161	3.62E+5
5	1.34E+6	0.161	2.15E+5
6	2.28E+6	0.161	3.68E+5
7	1.95E+6	0.161	3.15E+5
8	4.94E+3	1.27 x 0.161	1.01E+3
9	2.05E+5	0.161	3.30E+4
10	0	0	0.00E+0

Table 6-5 SUMMARY OF 1998 LEVEL II UPDATE FOR CNS

CNS Release Bin by Containment Failure Mode	Description	CNS Release Frequency	Percent Contribution
Containment Isolation Failure	Primary containment isolation failure or containment bypass (e.g., BOC, ISLOCA). Release pathway modeled through drywell.	7.39E-8	0.56%
CFE	Early containment failure from fuel-coolant interaction, drywell shell melt-through, overpressure due to vessel breach blowdown, or overpressure prior to core damage during loss of injection accidents (e.g., PSS, ATWS), Release pathway may be through drywell or wetwell, depending on phenomenon	5.49E-6	41.47%
RPV Vent	RPV vent successfully opened to main condenser during containment flooding process.	1.18E-6	8.91%
Hard Pipe Vent	Proceduraly directed use of hard-pipe containment vent (either early, before vessel breach or late, after vessel breach), and no other containment release pathways exist.	1.56E-6	11.78%
Sump Melt	Containment sump melt-through (w/o coincident containment overtemperature failure)	3.69E-7	2.79%
Overtemperature	Overtemperature failure of containment structure (release modeled through head seals)	3.96E-6	29.91%
Overpressure	Overpressure failure of containment structure (release pathway may be through drywell or wetwell)	9.92E-9	0.07%
No Containment Failure	No containment failures or procedurally directed containment release pathways (i.e., RPV vent, hard-pipe vent) occur.	5.96E-7	4.50%
Total		1.32E-5	100.00%

Table 6-6 SUMMARY OF CNS RELEASE FREQUENCY BY CONTAINMENT FAILURE MODE

CNS Release Bin by Containment Failure Mode	Description	CNS Release Frequency
Containment Isolation Failure	Primary containment isolation failure or containment bypass (e.g., BOC, ISLOCA). Release pathway modeled through drywell.	7.26E-8
CFE	Early containment failure from fuel-coolant interaction, drywell shell melt-through, overpressure due to vessel breach blowdown, or overpressure prior to core damage during loss of injection accidents (e.g., PSS, ATWS), Release pathway may be through drywell or wetwell, depending on phenomenon	5.39E-6
RPV Vent	RPV vent successfully opened to main condenser during containment flooding process.	1.16E-6
Hard Pipe Vent	Procedurally directed use of hard-pipe containment vent (either early, before vessel breach or late, after vessel breach), and no other containment release pathways exist.	1.53E-6
Sump Melt	Containment sump melt-through (w/o coincident containment overtemperature failure)	3.62E-7
Overtemperature	Overtemperature failure of containment structure (release modeled through head seals)	3.89E-6
Overpressure	Overpressure failure of containment structure (release pathway may be through drywell or wetwell)	9.74E-9
No Containment Failure	No containment failures or procedurally directed containment release pathways (i.e., RPV vent, hard-pipe vent) occur.	5.85E-7
Total		1.30E-5

Table 6-7 EPRI CONTAINMENT FAILURE CLASSIFICATIONS

Class	Description
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.
2	Containment isolation failures include accidents for which there is a failure to Isolate the containment.
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.
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 potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
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.
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).
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
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.

Table 6-8 MAPPING OF PEACH BOTTOM ACCIDENT PROGRESSION BINS TO EPRI ACCIDENT CLASS 8

EPRI Accident Classes	Description	Peach Bottom Collapsed Accident Progression Bin (APB)
1	No Containment Failure	8
2	Large Isolation Failures (Fail to Close)	3
3A	Small Isolation Failures (Liner Breach)	
3B	Large Isolation Failures (Liner Breach)	
4	Small Isolation Failures (Fail to Seal - Type B)	
5	Small Isolation Failures (Fail to Seal - Type C)	
6	Other Isolation Failures (e.g., dependent failures)	
7	Failures induced by Phenomena (early and late)	1,2,3,4,5,6,7,9
8	Bypass (Interfacing Systems LOCA)	3

Table 6-9 MAPPING OF CNS CONTAINMENT FAILURE MODES TO EPRI RELEASE CLASSES

CNS Release Bin by Containment Failure Mode	Description	EPRI Accident Class
Containment Isolation Failure	Primary containment isolation failure or containment bypass (e.g., BOC, ISLOCA). Release pathway modeled through drywell.	2,8
CFE	Early containment failure from fuel-coolant interaction, drywell shell melt-through, overpressure due to vessel breach blowdown, or overpressure prior to core damage during loss of injection accidents (e.g., PSS, ATWS), Release pathway may be through drywell or wetwell, dpending on phenomenon	7
RPV Vent	RPV vent successfully opened to main condenser during containment flooding process.	7
Hard Pipe Vent	Procedurally directed use of hard-pipe containment vent (either early, before vessel breach or late, after vessel breach), and no other containment release pathways exist.	7
Sump Melt	Containment sump melt-through (w/o coincident containment overtemperature failure)	7
Overtemperature	Overtemperature failure of containment structure (release modeled through head seals)	7
Overpressure	Overpressure failure of containment structure (release pathway may be through drywell or wetwell)	7
No Containment Failure	No containment failures or procedurally directed containment release pathways (i.e., RPV vent, hard-pipe vent) occur.	1

6.3 Conditional Probability of ILRT Failure (Small and Large)

The ILRT can detect a number of failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces. The proposed ILRT test interval extension may influence the conditional probability associated with the ILRT failure. To ensure that this effect is properly accounted for, the Class 3 Accident Class is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

Data presented in NUREG-1493 [4] was used to calculate the probability that a liner leak will be large (Event Class 3b). The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Because $21L_a$ does not constitute a large release, no releases have occurred based on the 144 ILRTs reported in NUREG-1493 [4].

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the χ^2 distribution. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [25]. The χ^2 distribution is really a family of distributions, which range in shape from exponential to normal. Each distribution is identified by the degrees of freedom, ν. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the χ^2 distribution can be calculated as χ^2_{95} ($\nu = 2n + 2$) / 2N, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and χ^2_{95} (2) = 5.99, the 95th percentile estimate of the probability of a large leak is calculated as 5.99 / (2 * 144) = 0.021.

To calculate the probability that a liner leak will be small (event Class 3a), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of $1.0L_a$. However, of these "failures" only 4 were found by an ILRT; the others were found by Type B and C testing on errors in test alignments. Therefore, out of the 144 ILRTs, 4 failures were categorized as "small releases". Similar to the event Class 3b probability, the estimated failure probability for small release is found by using the χ^2 distribution. The χ^2 distribution is calculated by n = 4 (number of small leaks) and N = 144 (number of events) which yields a. χ^2_{95} (10) = 18.3070. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070 / (2 * 144) = 0.064.

Using the methodology discussed above is conservative compared to the typical mean estimates used for PRA analysis. For example, the mean probability of a Class 3a failure would be the (number of failures) / (number of tests) or 4/144 = 0.03 compared with 0.064 used here.

6.4 Impact of Extension on Leak Detection Probability

The NRC in NUREG-1493 [4] has determined from a review of operating experience data that only 3% of the ILRT failures were found which local leakage-rate testing could not and did not detect. In NUREG-1493 [4], it is noted that based on a review of leak rate testing experience, a small percentage (3%) of leakages that exceed current requirements are detectable only by Type A testing (ILRT). Further, in NUREG-1493 it is noted that the leakage rates observed in these few Type A test failures were only marginally above currently prescribed limits and could be characterized by a leakage rate of about two times the allowable.

Also in NUREG-1493 [4], it was assumed that the characteristic magnitude of leakages detectable only by ILRTs would not change, but the probability of leakage would change due to the longer intervals between tests. The change in probability was 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 (3yrs/2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10yrs/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. However, since ILRTs have been demonstrated to improve the residual leak detection by only 3%, the interval change noted above would only lead to about a 10% (3.33 x 3%) non-detection leak probability. It is assumed that LLRT will continue to provide leak detection for 97% of leakages. Correspondingly, an extension of the ILRT interval to fifteen years is estimated to result in approximately a 15% (7.5/1.5x3%) non-detection probability of a leak. These are approximations assumed by the NRC and EPRI because the current 3 ILRTs in 10 years would have a T/2 = 1.67 years instead of 1.5 years.

Therefore, the failure rate of ILRTs for which the LLRTs do not provide adequate backup is 0.03/1.5 year average detection time. Applying a constant failure rate model, the failure probability of ILRTs, P_f, can be estimated as follows:

<u>For 3 Year Interval</u>

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{3yr}{2}\right) = 0.03$$

<u>For 10 Year Interval</u>

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{10yr}{2}\right) = 0.10$$

For 15 Year Interval

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{15yr}{2}\right) = 0.15$$

EPRI has previously interpreted this to mean that the failure to detect probability values is as follows:

ILRT FAILURE TO DETECT PROBABILITY

ILRT Interval	EPRI Assessment [2]	IP3 [14]	Constant Failure Rate Model
3 yr	0.03	0.03	0.03
10 yr	0.13	0.13	0.10
15 yr	NA	0.18	0.15

In addition, IP3 [14] has used this same estimate of changes in detection probability in a submittal to extend the ILRT interval on a one-time basis. The IP3 request for a one-time ILRT extension was approved by the NRC on April 17, 2000 (TAC No. MB0178) [21].

The analysis included in this report follows the precedence set by the EPRI report and the IP3 analysis. The use of the constant failure rate model is conservatively represented by the assumed "failure to detect" probabilities used by EPRI and in the IP3 submittal.

7. Assumptions

The following assumptions and ground rules were used in the engineering evaluation:

- The CNS Level I and Level II internal events PRA model provides representative results for the analysis. The CNS Level I does not include fire, floods, and shutdown events. However, a bounding sensitivity analysis is performed to assess the impact of external events.
- It is appropriate to use the CNS internal events PRA model as a gauge to effectively describe the risk changes 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, floods and shutdown events were to be included in the calculations.

- An evaluation of the risk trade-off impact of performing the ILRT during shutdown is addressed using the generic results from EPRI TR 105189 [10].

- Dose results for the containment failures modeled in the PRA can be characterized by the NUREG/CR-4551 population dose results [5] with corrections for CNS-specific population density and ILRT leakage criteria.

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized on Table 6-7.

- The maximum containment leakage for Class 1 sequences is 1.0 L_a. Class 3 accounts for increased leakage due to Type A inspection failures.

- The maximum containment leakage for Class 3a sequences is 10 L_a, based on the previously approved methodology [14, 21, 23].

- The maximum containment leakage for Class 3b sequences is 35 L_a, based on the previously approved methodology [14, 21, 23].

- The impact on population doses from Interfacing System Loss of Coolant Accidents (ISLOCAs) 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 ISLOCA contribution to population dose is fixed, no changes to the conclusions of this analysis will result from this assumption.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal. Containment isolation valves that fail to close during an accident and in response to a containment isolation signal are calculated on a CNS-specific basis and made part of the overall population dose and LERF calculations.

8. Application of EPRI TR-104285 Methodology

The application of the EPRI TR-104285 approach is based on EPRI-TR-105189 [10] and previous risk assessment submittals on this subject [14]. The approach has established a clear process for the calculation and presentation of results.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 8-1 lists these accident classes.

The analysis performed examined CNS specific accident sequences in which the containment either remains intact or is impaired. Specifically, the break down of the severe accident contribution 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).

- Accident sequences involving containment bypass (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.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the applicable eight accident classes presented in Table 8-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

Step 3 - Evaluate the risk impact of extending Type A test interval from 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine Impact on the Conditional Containment Failure Probability (CCFP).

Step 1 - Quantify the Base-line Risk in Terms of Frequency per Reactor Year

As discussed in Section 6.2 and as shown on Table 6-4, the population doses for the Peach Bottom Plant in NUREG/CR-4551 [5] were used to develop the population doses for CNS. The population doses for the ten Peach Bottom Accident Progression Bins were adjusted for population differences and ILRT leakage differences to obtain the equivalent CNS doses. As shown on Table 6-8, the ten Peach Bottom Accident Sequence Bins were mapped to the EPRI accident classes.

As discussed in Section 6.2 and as shown on Table 6-5, the 1998 update of the CNS Level II analysis was used to quantify percent contributions for radiological release categories based on containment failure mode and Revision 96b of the CNS PRA Model was used to quantify frequencies for the radiological release categories for based on CNS containment failure modes. As shown on Table 6-9, the CNS containment failure modes were mapped to the EPRI accident classes.

The extension of the Type A test 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 are included in the model. Specifically, a simplified model based on NUREG-1493 results is used to predict the likelihood of having a small/large breach in the containment liner that is undetected by the Type A ILRT test. These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]. The Class 3 leakage includes the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes, event Class-3a (small breach) and event Class-3b (large breach) were considered to ensure proper representation of available data.

After including the respective "large" and "small" liner breach leak rate probabilities (Classes 3a and 3b), the eight severe accidents class frequencies were developed consistent with the definitions in Table 8-1 and described below.

<u>Class 1 Sequences</u>

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 for these sequences is 5.85E-7/year based on the value in Table 6-6 and the mapping relationship in Table 6-9. After all accident class frequencies (Classes 2 through 8) were developed, frequencies for Classes 3A, 3B, and 6 were summed. This was then subtracted from the total 5.85E-7 to obtain the Class 1 frequency of "No Containment Failure" of 5.30E-7/yr. For this analysis, the associated maximum containment leakage for this group is $1.0L_a$, consistent with an intact containment evaluation.

Class 2 Sequences.

This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. These sequences are dominated by failure-to-close of large containment isolation valves. The frequency per year for these sequences is 7.26E-8/year based on the value on Table 6-6 and the mapping relationship on Table 6-9.

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 ($2.0L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.064 [see Section 6.3]

$PROB_{Class\ 3b}$ = probability of large pre-existing containment liner leakage
= 0.021 [see Section 6.3]

CLASS_3a_FREQUENCY = 0.064 * 5.85E-7/year = 3.74E-8/year

CLASS_3b_FREQUENCY = 0.021 * 5.85E-7/year = 1.23E-8/year

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is $35L_a$. These assignments are consistent with the Indian Point 3 ILRT submittal [14] which was approved by the NRC [21]. For 10-yr and 15-yr test intervals, there is a likelihood that corrosion related containment leakage may not be detected. Therefore, the baseline frequency for Class 3b sequences is increased by a factor of 1.000196 to account for undetected corrosion related containment leakage. (Appendix A presents the basis and supporting calculations). Note that this factor is conservatively based on a test interval increase from 3 years to 15 years and is used for the 10-year and 15-year cases.

Class 4 Sequences.

This group consists of all core damage accident progression bins for which a 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 containment isolation failure-to-seal of Type C test components. 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.

This group is similar to Class 2, and the low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the fact that this failure class is not impacted by Type A testing, a screening value is considered appropriate for this low probability failure mode. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i,e, $PROB_{class6}>0$), a conservative screening value of 4.0E-4 will be used to evaluate this class.

The frequency per year for these sequences is determined as follows:

$CLASS\_6\_FREQUENCY = PROB_{largeT\&M} * CDF$

Where:

$PROB_{largeT\&M}$ = random large containment isolation failure probability due to valve misalignment is estimated using NUREG/CR-1278.

= 4.0E-4

CLASS_6_FREQUENCY = 4.0E-4 * 1.30E-5/year
= 5.20E-9/year

For this analysis the associated containment leakage for this group is represented by the direct release from containment, i.e., Class 2 consequences are assigned.

Class 7 Sequences.

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., direct containment heating, melt-through, over-pressure). The baseline frequency per year for these sequences is 1.23E-5/year and is based on the values on Table 6-6 and the mapping relationship on Table 6-9. The mapped CNS bin frequencies are summed to obtain the EPRI Class 7 frequency.

Class 8 Sequences.

This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency per year for these sequences is

0.00E+/year and is based on the values on Table 6-6 and the mapping relationship on Table 6-9. The frequency for EPRI Class 8 was set to zero because it is already included as part of EPRI Class 2. This is acceptable since the population doses for EPRI Classes 2 and 8 are determined to be identical (See Table 8-3).

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 definition of Accident Classes defined in EPRI-TR-104285. Table 8-2 summarizes these accident frequencies by Accident Class.

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

In Section 6.2, a release analysis was performed to estimate the person-rem doses to the population, within a 50-mile radius from CNS. The releases are based on NUREG/CR-4551 [5] with adjustments for site-specific population and ILRT test leakage differences. The population doses for the EPRI classes were calculated using the population doses in Table 6-4 and the mapping relationships in Table 6-8.

Class 1 = 1.01E+3 person-rem (at $1.0L_a$)

Class 2 = 4.78E+5 person-rem

Class 3a = 1.01E+3 person-rem x $10L_a$ = 1.01E+4 person-rem

Class 3b = 1.01E+3 person-rem x $35L_a$ = 3.54E+4 person-rem

Class 4 = Not analyzed (Assigned a zero value)

Class 5 = Not analyzed (Assigned a zero value)

Class 6 = 4.78E+5 person-rem (Assumed a Class 2 release)

Class 7 = 3.64E+5 person-rem (sum of the population dose risk for Release Categories 1, 2, 3, 4, 5, 6, 7 and 9 on Table 6-4 divided by the NUREG/CR-4551 frequencies for Release Categories 1, 2, 3, 4, 5, 6, 7 and 9 on Table 6-2)

Class 8 = 4.78E+5 person-rem (Assumed a Class 2 release)

The population dose estimates derived for use in the risk evaluation are summarized in Table 8-3.

The above results, when combined with the results presented in Table 8-2, yield the baseline mean consequence measures for each accident class. These results are presented in Table 8-4.

The total dose per year is compared with the other sites as shown below:

Plant	Annual Dose (Person-Rem/yr)	Reference
Indian Point 3	14.515	14
Peach Bottom	6.2	15
Crystal River	1.4	16
CNS	4.5268	Table 8-4

Based on the risk values from Table 8-4, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 (i.e., the Class affected by the ILRT interval change) is as follows:

$$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS\ 3b_{BASE})/\ Total_{BASE}] \times 100$$

Where:
$CLASS3a_{BASE}$= Class 3a person-rem/year = 3.79E-4 person-rem/year [Table 8-4]
$CLASS3b_{BASE}$= Class 3b person-rem/year = 4.35E-4 person-rem/year [Table 8-4]
$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 4.52677 person-rem/yr [Table 8-4]

$$\%Risk_{BASE} = [(3.79E\text{-}4 + 4.35E\text{-}4)/4.52677] \times 100$$
$$\%Risk_{BASE} = 0.018\%$$

Step 3 – Evaluate Risk Impact of Extending Type A Test Interval from 10-to-15 Year

According to NUREG-1493 [4], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10-years will increase the average time that a leak detectable only by an ILRT goes undetected from 1.5 years to 5 years. The average time for failure to detect is calculated using the approximation ½ λT where T is the Test interval and λ, the leakage failure rate, is (3%)/1.5 year. If the test interval is extended to 1 in 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 7.5 years (1/2 * 15 years.). Because ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% undetectable rate in the overall probability of leakage ½ * (3% /1.5 years) * 10 years.

This value is determined by multiplying 3% and the ratio of the average time for non-detection for the increased ILRT test interval to the baseline average time for non-detection. For a 15-yr-test interval, the result is a 15% overall probability of leakage (i.e., ½ * (3% /1.5 yrs) * 15 years). Thus, increasing the ILRT test interval from 10 years to 15 years translates into a 5% increase in the overall leakage probability.

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 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1 (7% increase [10%-3%] which is approximated here as a factor of 1.1 consistent with the approach used by Indian Point 3 [14]). Specifically, there is a factor of 1.1 increase in the Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 10 yrs. (See Section 6.4)

Risk Impact of Corrosion Related Leakage due to Increase to 15-year Test Interval

Increasing the test interval from 3 to 15 years may reduce the chance of detecting corrosion related leakage. The likelihood of not detecting corrosion related leakage due to increased test interval from 3 to 15 years is calculated to be 0.0207%. Details of this calculation are provided in Appendix A. The calculation assumes that the total containment surface area below the containment spring-line (i.e., the dome-cylinder interface) can be exposed to corrosion. The increased likelihood of corrosion-related leakage is assumed to increase LERF contributions from EPRI Class 3B by a factor of 1.000207. This factor is applied to both 10-year and 15-year test interval calculations.

The results of this calculation are presented in Table 8-5. Based on the Table 8-5 values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class 3 is computed as follows:

$$(\%Risk_{10}) = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] \times 100$$

Where:
$CLASS3a_{10}$ = Class 3a person-rem/year = 4.17E-4 person-rem/yr [Table 8-5]
$CLASS\ 3b_{10}$ = Class 3b person-rem/year = 4.79E-4 person-rem/yr [Table 8-5]
$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 4.52685 person-rem/yr [Table 8-5]

$\%Risk_{10}$ = [(4.17E-4 + 4.79E-4) / 4.52685] x 100
$\%\ Risk_{10}$ = 0.02%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.02%.

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is computed as follows:

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 4.52677 person-rem/yr [Table 8-4]
$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 4.52685 person-rem/yr[Table 8-5]

$\Delta\%Risk_{10} = [(4.52685 - 4.52677) / 4.52677] \times 100.0$
$\Delta\%Risk_{10} = 0.002\%$

Therefore, the increase in risk due to the change in ILRT test frequency from three-in-ten years to 1-in-ten-years is 0.002%.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-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 15 percent or 1.15 consistent with previously approved method [14, 21]. Specifically, there is a factor of 1.15 increase in Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 15 yrs. (See Section 4.4) The results for this calculation are presented in Table 8-6.

Based on the values from Table 8-6, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$\%Risk_{15} = [(CLASS3a_{15} + CLASS3b_{15}) / TOTAL_{15}] \times 100$

Where:
$CLASS3a_{15}$ = Class 3a person-rem/year = 4.36E-4 person-rem/year [Table 8-6]
$CLASS3b_{15}$ = Class 3b person-rem/year = 5.01E-4 person-rem/year [Table 8-6]

$TOTAL_{15}$ = Total person-rem/yr for 15-year interval
= 4.52689 person-rem/yr [Table 8-6]

$\%Risk_{15} = [(4.36E-4 + 5.01E-4)/4.52689] \times 100$
$\%Risk_{15} = 0.021\%$

Therefore, the Total 15-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.021%.

The percent increase on the total integrated plant risk when the ILRT is extended from 10 years to 15 years is computed as follows:

$\%TOTAL_{10\text{-}15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$

Where:
$TOTAL_{10}$ = Total person-rem/year for 10-year interval
= 4.52685 person-rem/year [Table 8-5]

$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 4.52689 person-rem/year [Table 8-6]

$\%TOTAL_{10\text{-}15} = [(4.52689 - 4.52685) / 4.52685] \times 100$
$\%TOTAL_{10\text{-}15} = 0.0009\%$

Therefore, the percent increase in total plant risk for these accident sequences, based on going from a 10-year ILRT interval to a 15-year interval, as influenced by Type A testing, is only 0.0009%.

The percent risk increase ($\Delta Risk_{15}$) due to a fifteen-year ILRT interval over the baseline is computed as follows:

$\Delta Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100$

Where:
$TOTAL_{BASE}$ = Total person-rem/year for baseline interval
= 4.52677 person-rem/year [Table 8-4]
$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 4.52689 person-rem/year [Table 8-6]

$\%\Delta Risk_{15} = [(4.52689 - 4.52677) / 4.52677] \times 100$
$\%\Delta Risk_{15} = 0.003\%$

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from three-in-ten-years to one-per-fifteen years is 0.003%.

Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency (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. Class 3b is treated in this analysis as a potential LERF contributor. Class 3a is not treated as a "large" release. Therefore, for this evaluation, only Class 3b sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Other accident classes such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not part of LERF.

RG 1.174[3] 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 CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Baseline (3 Yr Test Interval) LERF

From Section 6.2, the baseline LERF frequency is:

$$LERF_{BASE} = 5.6E\text{-}7/year$$

LERF for 10-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is computed as follows:

$$\Delta LERF_{BASE\text{-}10} = CLASS3b_{10} - CLASS3b_{BASE}$$

Where:
$CLASS3b_{BASE}$ = 1.23E-8/yr [Table 8-4]
$CLASS3b_{10}$ = 1.35E-8/yr [Table 8-5]

$$\Delta LERF_{BASE\text{-}10} = 1.35E\text{-}8/yr - 1.23E\text{-}8/yr = 1.2E\text{-}9/yr$$

$$LERF_{10} = LERF_{BASE} + \Delta LERF_{BASE\text{-}10}$$

$$LERF_{10} = 5.600E\text{-}7/year + 1.2E\text{-}9/yr = 5.612E\text{-}7/yr$$

LERF for 15-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is computed as follows:

$$\Delta LERF_{BASE\text{-}15} = CLASS3b_{15} - CLASS3b_{BASE}$$

Where:
$CLASS3b_{BASE}$ = 1.23E-8/yr [Table 8-4]
$CLASS3b_{15}$ = 1.41E-8/yr [Table 8-6]

$$\Delta LERF_{BASE\text{-}15} = 1.41E\text{-}8/yr - 1.23E\text{-}8/yr = 1.8E\text{-}9/yr$$

$$LERF_{15} = LERF_{BASE} + \Delta LERF_{BASE\text{-}15}$$

$LERF_{15}$ = 5.600E-7/year + 1.8E-9/yr = 5.618E-7/year

The LERF increase ($\Delta LERF_{10\text{-}15}$) due to a 15-year ILRT over the 10-yr ILRT is as follows:

$\Delta LERF_{10\text{-}15} = CLASS3b_{15} - CLASS3b_{10}$

Where:
$CLASS3b_{10}$ = 1.35E-8/yr [Table 8-5]
$CLASS3b_{15}$ = 1.41E-8/yr [Table 8-6]

$\Delta LERF_{10\text{-}15}$ = 1.41E-8/yr – 1.35E-8/yr = 6.0E-10/yr

It should be noted that the calculated changes in LERF for all cases are less than the 1.0E-7/yr screening criterion in Reg. Guide 1.174 and represent a very small change in risk.

Step 5 – Determine Impact on the Conditional Containment Failure Probability

Another parameter that the NRC Guidance Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of 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 is calculated from the risk calculations performed in this analysis. The CCFP is "conditional" in that it identifies the probability of containment failure given that a severe accident (i.e., core damage) has occurred. Containment failure in this context includes all radionuclide release end states other than the intact state that do not involve containment bypass. Generally, this means non-bypass, non-Class 1 sequences.

Since the only classes that are increasing are Classes 3a and 3b, the change in CCFP can be calculated by the difference in these classes. The CCFP calculation for the base case ($CCFP_{BASE}$) is shown below:

The percent increase in CCFP ($\Delta\%CCFP_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is computed as follows:

$$\Delta\%CCFP_{BASE\text{-}10} = [((F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS3b\_BASE}))/CDF] \times 100$$
$$= [((4.12E\text{-}8 + 1.35E\text{-}8) - (3.75E\text{-}8 + 1.23E\text{-}8)) / 1.30E\text{-}5] \times 100$$
$$= 0.038\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is computed as follows:

$$\Delta\%CCFP_{BASE\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS3b\_BASE}))/CDF] \times 100$$
$$= [((4.31E\text{-}8 + 1.41E\text{-}8) - (3.75E\text{-}8 + 1.23E\text{-}8)) / 1.30E\text{-}5] \times 100$$
$$= 0.057\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{10\text{-}15}$) due to a 15-year ILRT over the 10-year ILRT is as follows:

$$\Delta\%CCFP_{10\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10}))/CDF] \times 100$$
$$= [((4.31E\text{-}8 + 1.41E\text{-}8) - (4.12E\text{-}8 + 1.35E\text{-}8)) / 1.30E\text{-}5] \times 100$$
$$= 0.019\%$$

This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

Results Summary

The following are the key results of the ILRT test interval extension risk analysis (using EPRI TR-104285 methodology):

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.018% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.020% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.002%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.021% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from the once-per-10-year frequency to once-per-15 years is 0.0009%.

6. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.003%.

7. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

8. The increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 1.2E-9/yr. This is considered to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

9. The increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 1.8E-9/yr. This is also considered to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

10. The increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 6.0E-10/yr. This is determined to be a very small using the acceptance guidelines of Reg. Guide 1.174.

11. The change in CCFP of less than 1% for both cases (when reducing test frequency to either once-per-10 or to once-per-15 years), is judged to be insignificant and reflects sufficient defense-in-depth.

Other significant results are summarized in Table 8-7.

Table 8-1 EPRI ACCIDENT CLASS DEFINITIONS

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)
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)

Table 8-2 EPRI ACCIDENT CLASS FREQUENCIES BASED ON CNS PRA - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	5.30E-7	4.08%
2	Large Isolation Failures (Fail to Close)	7.26E-8	0.56%
3A	Small Isolation Failures (Liner Breach)	3.75E-8	0.29%
3B	Large Isolation Failures (Liner Breach)	1.23E-8	0.09%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00%
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	0.04%
7	Failures induced by Phenomena (early and late)	1.23E-5	94.94%
8	Bypass (Interfacing Systems LOCA)	0.00E+0	0.00%
Total		**1.30E-5**	

Table 8-3 POPULATION DOSE ESTIMATES FOR CNS AT 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	1.01E+3
2	Large Isolation Failures (Failure to Close)	4.78E+5
3a	Small Isolation Failures (liner breach)	1.01E+4
3b	Large Isolation Failures (liner breach)	3.54E+4
4	Small Isolation Failures (Failure to seal-Type B)	0.00E+0
5	Small Isolation Failures (Failure to seal-Type C)	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	4.78E+5
7	Failures Induced by Phenomena	3.64E+5
8	Bypass (Interfacing System LOCA)	4.78E+5

Table 8-4 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.30E-7	1.01E+3	5.37E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	3.75E-8	1.01E+4	3.79E-4
3b	Large Isolation Failures (Liner Breach)	1.23E-8	3.54E+4	4.35E-4
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.52677**

Table 8-5 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.25E-7	1.01E+3	5.32E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	4.12E-8	1.01E+4	4.17E-4
3b	Large Isolation Failures (Liner Breach)	1.35E-8	3.54E+4	4.79E-4
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+	0.00E+	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.52685**

Table 8-6 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.23E-7	1.01E+3	5.29E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	4.31E-8	1.01E+4	4.36E-4
3b	Large Isolation Failures (Liner Breach)	1.41E-8	3.54E+4	5.01E-4
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.52689**

Table 8-7 SUMMARY OF RISK IMPACT OF TYPE A ILRT TEST FREQUENCIES - EPRI TR-104285

Risk Metric	Risk Impact (Baseline)	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.018% of total integrated value 8.14E-4 person-rem/yr	0.020% of total integrated value 8.96E-4 person-rem/yr	0.021% of total integrated value 9.37E-4 person-rem/yr
Total Integrated Risk	4.52677 person-rem/year	4.52685 person-rem/year	4.52689 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.002%	0.003%
Increase in LERF over Baseline	N/A	1.2E-9/yr	1.8E-9/yr
Percent Increase in CCFP over Baseline	N/A	0.038%	0.057%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.0009%
Increase in LERF over 10-yr ILRT	N/A	N/A	6.0E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.019%

9. Application of NEI Interim Guidance Methodology

9.1 Summary of Methodology

The results of the risk assessment performed using the methodology of EPRI TR-104285 [2], were provided in Section 8 of this document. In 2001 NEI recognized a need to update this methodology to support future risk-informed ILRT interval extension submittals. The methodology update focused on the following three particular areas.

1. The methodology for determining the overall probability of leakage resulting from extending surveillance intervals was revised. For an ILRT interval extension of 3 in 10 years to 1 in 10 years, the overall 10-year dose should have been calculated using an increased probability of an undetected leak (a leak detectable only by an ILRT that goes undetected due to the increased test interval) of 333.3% (increased by a factor of 3.33), as opposed to the 10% value used in the EPRI TR-104285 methodology. However, NEI also showed this methodology change to have a very small incremental risk contribution, since ILRTs only address a very small portion of the severe accident risk.

2. The methodology used to determine the frequencies of leakages detectable only by ILRTs (EPRI Classes 3a and 3b) was revised. Updated ILRT failure data was incorporated into the calculation of these containment failure classes. The guidance recommended use of a mean frequency calculation for the Class 3a distribution, and recommended the use of a Jeffery's non-informative prior distribution for the Class 3b distribution. The impact of this methodology change was to increase the probability of Class 3b releases. However, it was noted that no observed failure to date was even close in size to that necessary to cause a large release.

3. The updated guidance included provisions for utilizing NUREG/CR-4551 dose calculations, a necessary improvement to make the methodology usable for plants that do not have a Level-3 PRA.

Other improvements in the methodology include use of a simplified risk model (as opposed to the Containment Event Tree model used in EPRI TR-104285) to distinguish between those accident sequences that are affected by the status of the containment isolation system versus those that are a direct function of severe accident phenomena, and evaluation of the change in LERF by manipulating the probability of a pre-existing leak (for either Class 3a and 3b end states) of sufficient leak size to produce a large, early release.

9.2 Analysis Approach

This section presents the steps involved in performing the ILRT extension risk assessment based on the methodology of the 2001 NEI Interim Guidance.

The nine analysis steps identified in the NEI Interim Guidance are:

1. Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest. Note that Classes 4, 5, and 6 are not affected by changes in ILRT test frequency. Therefore, these classes are not considered in this assessment methodology.
2. Determine the containment leakage rates for applicable cases, 3a and 3b.
3. Develop the baseline population dose (person-rem, from the plant IPE, or calculated based on leakage) for the applicable accident classes.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).
   (Note: The method provides for use of the NUREG/CR-4551 population dose methods. If plant-specific values are available, they may be used. The net result is expressed as a percentile change.)
5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.
6. Determine the population dose rate for the new surveillance intervals of interest.
7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
8. Evaluate the risk impact in terms of LERF.
9. Evaluate the change in conditional containment failure probability.

Each of these steps are described in detail below. Note that this methodology builds upon the methodology of EPRI TR-104285. Therefore, most of the plant specific information necessary to perform the assessment using this methodology was presented in Sections 6 and 8 above (reference is made as necessary to the appropriate section in Section 6 or 8 for the development of the common information).

Step 1: Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

The baseline EPRI accident class frequencies used in the NEI methodology case are unchanged from those calculated in Sections 6 and 8 above, with the exceptions of the frequencies for EPRI categories 1 (No Containment Failure) and 3a (Small Containment Isolation Failures due to Liner Breach) and 3b (Large Containment Isolation Failures due to Liner Breach). As described above, the frequencies of leakages detectable only by ILRTs (EPRI Classes 3a and 3b) was revised. The NEI Interim Guidance included the results of additional, updated ILRT failure data (38 more

industry tests conducted since 1/1/1995). Adding these to the NUREG-1493 data (144 ILRTs) resulted in a total population of 182 tests. One more failure was added (due to construction debris from a penetration modification), resulting in a total of 5 failures over these 182 tests. The guidance recommended use of a mean frequency (5/182 = 0.027) for the Class 3a distribution, and recommended the use of a Jeffery's non-informative prior distribution for the Class 3b distribution:

$$\begin{aligned}\text{Failure Probability}_{3b} &= (\text{Number of Failures} + ½) / (\text{Number of Tests} + 1)\\ &= (0 + ½) / (182 + 1)\\ &= 0.0027\end{aligned}$$

Using these values, the calculation of the baseline Class 3a and 3b frequencies was performed by multiplying them to IPE Class 1 frequency as follows:

CLASS_3a_FREQUENCY = 0.027 * 5.85E-7/year = 1.58E-8/year

CLASS_3b_FREQUENCY = 0.0027 * 5.85E-7/year = 1.58E-9/year

In order to maintain the sum of the frequencies of the accident classes equal to the CDF, the NEI Interim Guidance specifies that the Class 1 frequency be adjusted for the Class 3 sequences. In addition, core damage sequences involving other containment isolation failures (pathways left 'open' following a plant post-maintenance tests, Class 6 sequences) were also removed from the Class 1 results. The baseline Class 1 frequency was determined as follows:

CLASS_1_FREQUENCY
= (IPE Class 1) - (Class 3a + Class 3b) – (Class 6)
= 5.85E-7/year – (1.58E-8/year + 1.58E-9/year)/yr – 5.20E-9/yr
= 5.63E-7/year

Table 9-1 below provides the CNS accident class frequencies that were used in the application of the NEI Interim Guidance methodology.

Step 2: Determine the containment leakage rates for applicable cases, 3a and 3b.

Step 3: Develop the baseline population dose (person-rem, from the plant IPE, or calculated based on leakage) for the applicable accident classes.

Step 4: Determine the population dose rate (person-rem/year) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).

Each of the calculations necessary for these steps were performed exactly as presented in Section 8. The resulting population dose rates for all accident classes are identical to that presented in Section 8, with the exception of Classes 1, 3a and 3b (the accident sequence frequencies of which were modified per the NEI guidance as described in Step 1 above). Table 9-2 provides the baseline results for the population dose rates by accident class.

The calculation of the baseline risk contribution from Class 3 (i.e., the class affected by the ILRT interval change) was also done consistent with the method presented in Section 8.2. Based on the risk values from Table 9-2, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 is as follows:

$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS\ 3b_{BASE})/\ Total_{BASE}] \times 100$

Where:
$CLASS3a_{BASE}$ = Class 3a person-rem/year = 1.60E-4 person-rem/year [Table 9-2]
$CLASS3b_{BASE}$ = Class 3b person-rem/year = 5.60E-5 person-rem/year [Table 9-2]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval
= 4.5262 person-rem/yr [Table 9-2]

$\%Risk_{BASE} = [(1.60E\text{-}4 + 5.60E\text{-}5)/\ 4.5262] \times 100$
$\%Risk_{BASE} = 0.005\%$

Steps 5, 6, and 7 are performed below the description for Step 7.

Step 5: Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

Step 6: Determine the population dose rate for the new surveillance intervals of interest.

Step 7: Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

The increase in the Class 3 leakage frequencies for the surveillance intervals of interest (10 years and 15 years) were computed using the same methodology used in Section 8 above, except that the overall 10-year dose was calculated using an increased probability of an undetected leak of 333.3% (increased by a factor of 3.33), as opposed to the 10% value (factor of 1.1) used in the EPRI TR-104285 methodology. Likewise, the overall 15-year dose was calculated using an increased probability of an undetected leak of 500% (increased by a factor of 5.0). As described in the NEI Interim Guidance, increasing the test interval from 3 in 10 years to 1 in 10 years increases the average time that a leak (detectable only by an ILRT) goes undetected from 18 (3yrs/2) to 60 (10 yrs/2) months. This is a factor of 60/18=3.333. By the same logic, increasing the test interval from 3 in 10 years to 1 in 15 years increases the average time that a leak goes undetected from 18 (3yrs/2) to 90 (15 yrs/2) months, a factor of 90/18 = 5.0.

The increase in Class 3B frequency due to undetected corrosion-related leakage, calculated in Appendix A, was included in the calculation as described in Section 8 above.

Tables 9-3 and 9-4 provide the results of the population dose rate calculations for the cases where the ILRT interval is extended to 10 years and 15 years, respectively.

Based on the risk values from Tables 9-3 and 9-4, the percent risk contribution for Class 3 over the two proposed ILRT extension intervals ($\%Risk_{10}$ and $\%Risk_{15}$) was calculated as follows:

$CLASS3a_{10}$ = Class 3a person-rem/year = 5.33E-4 person-rem/year [Table 9-3]
$CLASS\ 3b_{10}$ = Class 3b person-rem/year = 1.87E-4 person-rem/year [Table 9-3]

$CLASS3a_{15}$ = Class 3a person-rem/year = 8.00E-4 person-rem/year [Table 9-4]
$CLASS\ 3b_{15}$ = Class 3b person-rem/year = 2.80E-4 person-rem/year [Table 9-4]

$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 4.5267 person-rem/yr [Table 9-3]
$TOTAL_{15}$ = Total person-rem/yr for 15-year interval = 4.5270 person-rem/yr [Table 9-4]

$\%Risk_{10}$ = [(5.33E-4 + 1.87E-4) / 4.5267] x 100
$\%Risk_{10}$ = 0.016%

$\%Risk_{15}$ = [(8.00E-4 + 2.80E-4) / 4.5270] x 100
$\%Risk_{15}$ = 0.024%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.016%for the ILRT interval extension to 1 in 10 years, and 0.024% for the ILRT interval extension to 1 in 15 years.

The percent risk increase ($\Delta\%Risk$) for each ILRT extension case over the baseline case is computed as follows:

$\Delta\%Risk_{10}$ = [($Total_{10}$ – $Total_{BASE}$) / $Total_{BASE}$] x 100.0
$\Delta\%Risk_{15}$ = [($Total_{15}$ – $Total_{BASE}$) / $Total_{BASE}$] x 100.0

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 4.5262 person-rem/yr [Table 9-2]
$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 4.5267 person-rem/yr [Table 9-3]
$TOTAL_{15}$ = Total person-rem/yr for 15 yr ILRT interval = 4.5270 person-rem/yr [Table 9-4]

$\Delta\%Risk_{10}$ = [(4.5267 – 4.5262) / 4.5262] x 100.0
$\Delta\%Risk_{10}$ = 0.011%

$\Delta\%Risk_{15}$ = [(4.5270 – 4.5262) / 4.5262] x 100.0
$\Delta\%Risk_{15}$ = 0.018%

The percent risk increase ($\Delta\%Risk$) for ILRT extension from 1 in 10 years to 1 in 15 years is computed as follows:

$\Delta\%Risk_{15\text{-}10}$ = [($Total_{15}$ – $Total_{10}$) / $Total_{10}$] x 100.0

$\Delta\%Risk_{15\text{-}10} = [(4.5270 - 4.5267) / 4.5267] \times 100.0$
$\Delta\%Risk_{15\text{-}10} = 0.007\%$

Therefore, the increase in risk due to changing the ILRT test interval of three-in-ten years to 1-in-ten-years is 0.011%, while the increase due to changing the ILRT test interval of three-in-ten years to a 1-in-15 year test interval is 0.018%. The increase due to changing the ILRT test interval of 1-in-10 years to a 1-in-15 year test interval is 0.007%

Step 8: Evaluate the risk impact in terms of LERF.

Baseline (3 Yr Test Interval) LERF

From Section 6.2, the baseline LERF frequency is:

$LERF_{BASE} = 5.6E\text{-}7/year$

LERF for 10-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is as follows:

$\Delta LERF_{BASE\text{-}10} = CLASS3b_{10} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE} = 1.58E\text{-}9/yr$ [Table 9-2]
$CLASS3b_{10} = 5.27E\text{-}9/yr$ [Table 9-3]

$\Delta LERF_{BASE\text{-}10} = 5.27E\text{-}9/yr - 1.58E\text{-}9/yr = 3.69E\text{-}9/yr$

$LERF_{10} = LERF_{BASE} + \Delta LERF_{BASE\text{-}10}$

$LERF_{10} = 5.600E\text{-}7/year + 3.69E\text{-}9/yr = 5.637E\text{-}7/yr$

LERF for 15-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 15-year ILRT over the baseline is as follows:

$\Delta LERF_{BASE\text{-}15} = CLASS3b_{15} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE} = 1.58E\text{-}9/yr$ [Table 9-2]
$CLASS3b_{15} = 7.90E\text{-}9/yr$ [Table 9-4]

$\Delta LERF_{BASE-15}$ = 7.90E-9/yr – 1.58E-9/yr = 6.32E-9/yr

$LERF_{15} = LERF_{BASE} + \Delta LERF_{BASE-15}$

$LERF_{15}$ = 5.600E-7/year + 6.32E-9/yr = 5.663E-7/year

The LERF increase ($\Delta LERF_{10-15}$) due to a 15-year ILRT over the 10-yr ILRT is as follows:

$\Delta LERF_{10-15} = CLASS3b_{15} - CLASS3b_{10}$

Where:
$CLASS3b_{10}$ = 5.27E-9/yr [Table 9-3]
$CLASS3b_{15}$ = 7.90E-9/yr [Table 9-4]

$\Delta LERF_{10-15}$ = 7.90E-9/yr - 5.27E-9/yr = 2.63E-9/yr

It should be noted that the calculated changes in LERF for all cases are less than the 1.0E-7/yr screening criterion in RG 1.174 and represent a very small change in risk.

Step 9: Evaluate the change in conditional containment failure probability.

The assessment of conditional containment failure probability (CCFP) for each of the cases (base, 10-year ILRT interval extension, 15-year ILRT interval extension) is performed in accordance with the NEI Interim Guidance methodology [23].

The CCFP calculation for the base case ($CCFP_{BASE}$) is shown below [23]:

$CCFP_{BASE}$ = 1 – (Intact Containment $Frequency_{BASE}$/Total CDF)
= {1 – (Class 1_{BASE} + Class $3a_{BASE}$)/CDF}*100
= {1 – (5.63E-7 + 1.58E-8)/1.30E-5}*100
= 99.55%

The CCFP calculation for the ILRT extension cases ($CCFP_{10}$ and $CCFP_{15}$) is performed in a similar manner:

$CCFP_{10}$ = 1 – (Intact Containment $Frequency_{10}$/Total CDF)
= {1 – (Class 1_{10} + Class $3a_{10}$)/CDF}*100
= {1 – (5.22E-7 + 5.27E-8)/1.30E-5}*100
= 95.579%

$CCFP_{15}$ = 1 – (Intact Containment $Frequency_{15}$/Total CDF)
= {1 – (Class 1_{15} + Class $3a_{15}$)/CDF}*100
= {1 – (4.93E-7 + 7.90E-8)/1.30E-5}*100

$= 95.6\%$

The percent increase in CCFP ($\Delta\%CCFP_{BASE-10}$) from a 10-year to a 15-year ILRT is computed as follows:

$$\Delta\%CCFP_{BASE-10} = CCFP_{10} - CCFP_{BASE}$$
$$= 95.58\% - 95.55\%$$
$$= 0.03\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE-15}$) due to a 15-year ILRT over the baseline is as follows:

$$\Delta\%CCFP_{BASE-15} = CCFP_{15} - CCFP_{BASE}$$
$$= 95.6\% - 95.55\%$$
$$= 0.05\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{10-15}$) due to a 15-year ILRT over the 10-year ILRT is as follows:

$$\Delta\%CCFP_{10-15} = CCFP_{15} - CCFP_{10}$$
$$= 95.6\% - 95.58\%$$
$$= 0.02\%$$

This change in CCFP is judged to be insignificant and reflects sufficient defense-in-depth.

9.3 Results Summary

The following are the key results of the ILRT test interval extension risk analysis based on the NEI Interim Guidance Methodology:

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.005% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.16% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.011%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.024% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.018%.

6. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the RG 1.174 acceptance guidelines.

7. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 3.69E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

8. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 6.32E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

9. The risk increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 2.63E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

10. The change in CCFP of less than 1% for both cases, reducing test frequency to either once-per-10 or once-per-15 years, is judged to be insignificant and reflects sufficient defense-in-depth.

Other results are summarized in Table 9-5.

Table 9-1 EPRI ACCIDENT CLASS FREQUENCIES FOR CNS - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	5.63E-7	4.33%
2	Large Isolation Failures (Fail to Close)	7.26E-8	0.56%
3A	Small Isolation Failures (Liner Breach)	1.58E-8	0.12%
3B	Large Isolation Failures (Liner Breach)	1.58E-9	0.01%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00%
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	0.04%
7	Failures induced by Phenomena (early and late)	1.23E-5	94.94%
8	Bypass (Interfacing Systems LOCA)	0.00E+0	0.00%
Total		**1.30E-5**	

Table 9-2 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.63E-7	1.01E+3	5.69E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	1.58E-8	1.01E+4	1.60E-4
3b	Large Isolation Failures (Liner Breach)	1.58E-9	3.54E+4	5.60E-5
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5262**

Table 9-3 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.22E-7	1.01E+3	5.28E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	5.27E-8	1.01E+4	5.33E-4
3b	Large Isolation Failures (Liner Breach)	5.27E-9	3.54E+4	1.87E-4
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5267**

Table 9-4 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	4.93E-7	1.01E+3	4.99E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	7.90E-8	1.01E+4	8.00E-4
3b	Large Isolation Failures (Liner Breach)	7.90E-9	3.54E+4	2.80E-4
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5270**

Table 9-5 SUMMARY OF RISK IMPACT ON TYPE A ILRT FREQUENCY - NEI Interim Guidance

Risk Metric	(Baseline) Risk Impact	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.005% of total integrated value 2.16E-4 person-rem/yr	0.016% of total integrated value 7.20E-4 person-rem/yr	0.024% of total integrated value 1.08E-3 person-rem/yr
Total Integrated Risk	4.5262 person-rem/year	4.5267 person-rem/year	4.5270 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.011%	0.018%
Increase in LERF over Baseline	N/A	3.69E-9/yr	6.32E-9/yr
Percent Increase in CCFP over Baseline	N/A	0.03%	0.05%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.007%
Increase in LERF over 10-yr ILRT	N/A	N/A	2.63E-9/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.02%

10. Application of EPRI TR-1009325 Methodology

10.1 Summary of Methodology

EPRI TR-1009325 [30] is an update to EPRI TR-104285 [2] (which, in turn, was built upon the guidance of NUREG-1493 [4]) that includes the changes to the methodology included in the NEI Interim Guidance [23], plus additional enhancements that were obtained through an expert elicitation process. In addition, the methodology incorporates the results of NRC comments on various industry ILRT interval extension submittals. The expert elicitation was aimed at reducing the conservatisms associated with the various containment leakage methodologies available that were found to provide widely differing risk results when applied to the same problem. The methodology enhancements support relaxation of ILRT intervals up to 20 years (ILRT extension had only requested interval extensions up to 15 years).

The enhancements in TR-1009325 are generally in the following three areas:

1. Definition (in terms of the required resulting L_a leakage term) of the assumed containment leakage size that could lead to a large, early release (LERF), i.e., EPRI accident Class 3b. Whereas previous submittals assumed a very conservative leakage term (35 L_a) would have the potential to result in a LERF event, the methodology provides a basis for using a (still conservative) value of 100 L_a instead. For the smaller pre-existing leak (accident Class 3a) size, the previously used conservative value of 10 L_a was retained by the methodology.
2. Development of specific probabilities for pre-existing containment leakage sizes. This was done through the expert elicitation process. EPRI TR-1009325 states that this method provides a considerable improvement over the use of non-informative priors (as has been done in previous licensee submittals based on application of the previous EPRI TR-104285 methodology).
3. Consideration of the potential risk benefits associated with other containment inspections (non-ILRT) and potential indirect containment monitoring techniques that would provide indications of a containment leak (determination of the probability of leakage detection over an increased ILRT interval, again through use of the expert elicitation process).

Application of the EPRI TR-1009325 methodology generally produces results that indicate lower population dose risk than previous methodologies due to the reduction in the conservatisms noted above.

10.2 Analysis Approach

Implementation of the methodology of EPRI TR-1009325 is very similar to the implementation of the NEI Interim Guidance discussed in Section 9.2 (the steps required for the analysis identified in TR-1009325 are identical with those presented in the NEI Interim Guidance). The practical differences between the two analyses are in the inputs used for determining the leak size requirements for LERF categorization (EPRI Class 3b), and in the probability values applied to the assumed undetected leakage categories. Therefore, in this section, the calculation discussion focuses on the changes in these inputs only. The calculation details followed are identical to those shown for the NEI Interim Guidance (Section 9.2). The presentation of results in Section 10.3 mirrors that provided for the other two methodologies.

Step 1: Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

Step 1 was quantified as described in Section 9.3, except in the leakage size and probabilities determined for Class 3a and Class 3b accident sequences.

Licensee ILRT extension submittals based on previous methodologies (EPRI TR-104285, 2001 NEI Interim Guidance) relied upon statistical failure data updates using non-informative priors in order to determine the probability values for containment leakage identifiable only through ILRTs (particularly Class 3b). As the risk results are sensitive to the 3b values, the choice of statistical methodology applied was seen to produce a somewhat wide range of risk results. EPRI TR-1009325 used expert elicitation to develop a relationship between the size of potential containment leakage pathways, expressed as L_a, and the probability of occurrence. This methodology was seen as a considerable improvement over the use of non-informative priors.

A summary of the final results of the statistical analysis of the expert elicitation (leak size vs. probability) are given in Table 6-1 of EPRI TR-1009325. As stated in Section 10.1, for Class 3 leakage scenarios, the EPRI TR-1009325 methodology specifies the use of 10 L_a as a conservative upper bound leakage size for Class 3a sequences, and 100 L_a as a conservative upper bound leakage size for Class 3b sequences. From Table 6-1 of EPRI TR-1009325, the mean probability of occurrence for a 10 L_a (Class 3a) leak is 3.88E-3, and the mean probability of occurrence for a 100 L_a (Class 3b) leak is 2.47E-4. Using these values, the calculation of the baseline Class 3a and 3b distributions was performed as follows:

CLASS_3a_FREQUENCY = 3.88E-3 * 5.85E-7/year = 2.27E-9/year

CLASS_3b_FREQUENCY = 2.47E-4 * 5.85E-7/year = 1.45E-10/year

These values are about an order of magnitude lower than the values calculated in Sections 8.1 (TR-104285 methodology) and 9.2 (NEI Interim Guidance methodology).

The remainder of the Step 1 calculation follows the same process as that presented in Section 9.2 above.

Steps 2 – 9:

The process followed to complete Steps 2 – 9 for the EPRI TR-1009325 methodology was the same as that presented in Section 9.2. Tables 10-1 through 10-4 provide the interim results of the EPRI TR-1009325 methodology.

10.3 Results Summary

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis (as calculated in Section 10 – EPRI TR-1009325 Methodology):

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.0008% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.0028% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.0018%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.0042% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.0031%.

6. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

7. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 3.37E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

8. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 5.78E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

9. The risk increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 2.41E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

10. The change in CCFP of less than 1% for both cases, reducing test frequency to either once-per-10 or once-per-15 years, is judged to be insignificant and reflects sufficient defense-in-depth.

Other significant results are summarized in Table 10-5.

Table 10-1 EPRI ACCIDENT CLASS FREQUENCIES FOR CNS - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	5.78E-7	4.44%
2	Large Isolation Failures (Fail to Close)	7.26E-8	0.56%
3A	Small Isolation Failures (Liner Breach)	2.27E-9	0.02%
3B	Large Isolation Failures (Liner Breach)	1.45E-10	0.00%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00%
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	0.04%
7	Failures induced by Phenomena (early and late)	1.23E-5	94.94%
8	Bypass (Interfacing Systems LOCA)	0.00E+0	0.00%
Total		**1.30E-5**	

Table 10-2 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.78E-7	1.01E+3	5.85E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	2.27E-9	1.01E+4	2.30E-5
3b	Large Isolation Failures (Liner Breach)	1.45E-10	1.01E+5	1.46E-5
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5260**

Table 10-3 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.72E-7	1.01E+3	5.79E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	7.57E-9	1.01E+4	7.66E-5
3b	Large Isolation Failures (Liner Breach)	4.82E-10	1.01E+5	4.88E-5
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5262**

Table 10-4 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose at 50 Miles (person-rem)	Population Dose Rate at 50 Miles (person-rem/yr)
1	No Containment Failure	5.68E-7	1.01E+3	5.75E-4
2	Large Isolation Failures (Fail to Close)	7.26E-8	4.78E+5	3.47E-2
3a	Small Isolation Failures (Liner Breach)	1.14E-8	1.01E+4	1.15E-4
3b	Large Isolation Failures (Liner Breach)	7.23E-10	1.01E+5	7.31E-5
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+0	0.00E+0	0.00E+0
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+0	0.00E+0	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	5.20E-9	4.78E+5	2.49E-3
7	Failures induced by Phenomena (early and late)	1.23E-5	3.64E+5	4.49E+0
8	Bypass (Interfacing Systems LOCA)	0.00E+0	4.78E+5	0.00E+0
Total		**1.30E-5**		**4.5262**

Table 10-5 SUMMARY OF RISK IMPACT OF TYPE A ILRT TEST FREQUENCIES - EPRI TR-1009325

Risk Metric	Risk Impact (Baseline)	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.0008% of total integrated value 3.76E-5 person-rem/yr	0.0028% of total integrated value 1.25E-4 person-rem/yr	0.0042% of total integrated value 1.88E-4 person-rem/yr
Total Integrated Risk	4.5260 person-rem/year	4.5261 person-rem/year	4.5262 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.0018%	0.0031%
Increase in LERF over Baseline	N/A	3.37E-10/yr	5.78E-10/yr
Percent Increase in CCFP over Baseline	N/A	0.0026%	0.0044%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.0013%
Increase in LERF over 10-yr ILRT	N/A	N/A	2.41E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.0019%

11. External Event Impacts

External hazards were evaluated in the CNS Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Although the external event hazards in the CNS IPEEE were evaluated to varying levels of conservatism, the results of the CNS IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment. The proposed ILRT interval extension impacts plant risk in a limited way. Specifically, the probability of a pre-existing containment leak being the initial containment failure mode given a core damage accident is potentially higher when the ILRT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events. The spectrum of external hazards has been evaluated in the CNS IPEEE by screening methods with varying levels of conservatism. Therefore, it is not possible at this time to incorporate a realistic quantitative risk assessment of all external event hazards into the ILRT extension assessment. As a result, external events have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix B of this calculation. As can be seen from Appendix B, if the external hazard risk results of the CNS IPEEE are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years is estimated at 1.70E-7/yr. This increase is in the range of 1E-07/yr to 1E-06/yr, which is in Region II of the RG 1.174 LERF acceptability curve. However, based on the conservative nature of the sensitivity study, it is expected that a more detailed external event study would reduce the estimated increase in LERF to less than 1E-07/yr. For example, the sensitivity study counted the full estimated seismic CDF and full estimated fire CDF against the 3b frequency. Table 8-6 shows that the 3b frequency is 0.11% of the total Internal Events CDF for the 15 year ILRT Test Interval. It should be noted that Reference [B2] did not identify any unique containment vulnerabilities for the seismic and fire evaluations.

The acceptance guidelines for Region II of the RG 1.174 LERF acceptability curve are that it can be reasonably shown that the total LERF is less than 1E-05/yr and that the cumulative changes be tracked. The baseline LERF is 5.60E-07. Based the LERF increase calculated using the NEI Interim Guidance (i.e., 1.70E-07), the total LERF for the requested change is 7.30E-07/yr which meets the total LERF criterion.

12. Conclusions

This section provides the principal conclusions of the ILRT test interval extension risk assessments as reported for the following:

- Previous generic risk assessment by the NRC
- CNS-specific risk assessment for the at-power case, performed using three available methodologies (EPRI TR-104285, NEI Interim Guidance, and EPRI TR-1009325)
- General conclusions regarding the beneficial effects on shutdown risk

12.1 Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years results in 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 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, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment liner.

12.2 CNS Specific Risk Results

The findings for CNS confirm the general findings of previous studies on a plant specific basis, including severe accident category frequencies, the containment failure modes, the Technical Specification allowed leakage, and the local population surrounding the CNS station. Based on the results from Sections 8 through 11, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the RG 1.174 acceptance guidelines.

- RG 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 in CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is between 2.41E-10/yr and 2.63E-9/yr. Guidance in RG 1.174 defines very small changes in LERF as below 1E-7/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the CNS risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk by between 0.0009% and 0.007%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- The change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

- Incorporating external event results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the CNS ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

Table 12-1 below summarizes the CNS-specific results of this risk evaluation.

Table 12-1 OVERALL SUMMARY OF RISK IMPACT OF VARIOUS TYPE A ILRT TEST FREQUENCIES

Risk Metric	Risk Impact (Baseline)			Risk Impact (10-years)			Risk Impact (15-years)		
Calculation Methodology	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325
Class 3a and 3b Risk Contribution	0.018% of total integrated value 8.14E-4 person-rem/yr	0.005% of total integrated value 2.16E-4 person-rem/yr	0.0008% of total integrated value 3.76E-5 person-rem/yr	0.020% of total integrated value 8.96E-4 person-rem/yr	0.016% of total integrated value 7.20E-4 person-rem/yr	0.0028% of total integrated value 1.25E-4 person-rem/yr	0.021% of total integrated value 9.36E-4 person-rem/yr	0.024% of total integrated value 1.08E-3 person-rem/yr	0.0042% of total integrated value 1.88E-4 person-rem/yr
Total Integrated Risk	4.52677 person-rem/year	4.5262 person-rem/year	4.5260 person-rem/year	4.52685 person-rem/year	4.5267 person-rem/year	4.5261 person-rem/year	4.52689 person-rem/year	4.5270 person-rem/year	4.5262 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	N/A	N/A	0.002%	0.011%	0.0018%	0.003%	0.018%	0.0031%
Increase in LERF over Baseline	N/A	N/A	N/A	1.2E-9/yr	3.69E-9/yr	3.37E-10/yr	1.8E-9/yr	6.32E-9/yr	5.78E-10/yr
Percent Increase in CCFP over Baseline	N/A	N/A	N/A	0.038%	0.03%	0.0026%	0.057%	0.05%	0.0044%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	N/A	N/A	N/A	N/A	0.0009%	0.007%	0.0013%
Increase in LERF		N/A			N/A			2.63E-9/yr	

over 10-yr ILRT	N/A		N/A	N/A		N/A	6.0E-10/yr		2.41E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	N/A	N/A	N/A	N/A	0.019%	0.02%	0.0019%

12.3 Risk Trade-off

The performance of an ILRT occurs during plant shutdown and introduces some small residual risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real shutdown risk impacts associated with the setup and performance of the ILRT during shutdown operation [10]. While these risks have not been quantified for CNS, it is judged that there is a positive (yet un-quantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT test interval extension, there are, in fact, positive safety benefits that reduce the already small risk associated with the extension of the ILRT test interval.

13. References

1) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 1995.
2) EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA," August 1994.
3) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
4) NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995.
5) NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2, Main Report," December 1990.
6) NUREG-1273, "Technical Findings and Regulatory Analysis for Genetic Study Issue II.e.43 'Containment Integrity Check'," April 1988.
7) NUREG/CR-3539, ORNL/TM-8964, "Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory," April 1984.
8) NUREG/CR-4220, PNL-5432, "Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory," June 1985.
9) NUREG/CR-4330, PNL-5809, Vol. 2, "Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory," June 1986.
10) TR-105189, Final Report, "Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA," May 1995.
11) Individual Plant Examination Peach Bottom Atomic Power Station Units 2 and 3, Volumes 1 and 2 Philadelphia Electric Company, 1992.
12) DE-ACOG-87RL11313, "ALWR Severe Accident Dose Analysis," March 1989
13) Not Used.
14) Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 16, 2001.

15) Letter from J. A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DDPR-56, LAR 01-00430, dated May 30, 2001.
16) Letter from D. E. Young (Florida Power) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
17) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
18) NUREG/CR-3539, "Burns, T. J., Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory," April 1984.
19) WASH-1400, "United States Nuclear Regulatory Commission, Reactor Safety Study," October 1975.
20) Letter from SNC (H. L. Summer, Jr.) to USNRC dated July 26, 2000.
21) United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment (Amendment No. 206) Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
22) NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 14, Regarding Cooper Nuclear Power Plant," dated January 2004.
23) J. Haugh, John Gisclon, W. Parkinson, Ken Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Intervals", Rev. 3.1, EPRI, October 2001.
24) CNS Level II Update, 1998
25) Patrick D.T. O'Connor, "Practical Reliability Engineering", John Wiley & Sons, 2nd Edition, 1985.
26) Letter from S. B. Minahan (NPPD) to NRC "Risk-Informed Inspection Program (Relief Request RI-34)," March 11, 2004, ADAMS Accession No. ML040760812.
27) Letter from R. K. Edington (NPPD) to NRC "Response to Request for Additional Information Regarding Risk-Informed Relief Request RI-34," July 29, 2004, ADAMS Accession No. ML042160125.
28) United States Nuclear Regulatory Commission, Cooper Nuclear Station – "RE: Risk-Informed Inspection Program (Relief Request RI-34)," (TAC No. MC2359), December 9, 2001.
29) Version 1996b of the CNS Internal Events PRA
30) EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," December, 2003.

APPENDIX A – CONTAINMENT CORROSION ANALYSIS

EFFECT OF AGE-RELATED DEGRADATION ON RISK INFORMED/RISK IMPACT ASSESSMENT FOR EXTENDING CONTAINMENT TYPE A TEST INTERVAL

A.1. Purpose

The purpose of this calculation is to assess the effect of age-related degradation of the containment on the risk impact for extending the CNS Integrated Leak Rate Test (ILRT) or Containment Type A test) interval from ten to fifteen years.

A.2. Intended Use of Analysis Results

The results of this calculation will be used to indicate the sensitivity of the risk associated with the extension in the ILRT interval to potential age-related degradation of the containment shell to support obtaining NRC approval to extend the Integrated Leak Rate Test (ILRT) interval at CNS from 10 years to 15 years. This calculation actually evaluates the impact of extending the interval from 3 years to 15 years.

A.3. Technical Approach

The present analysis shows the sensitivity of the results of the assessment of the risk impact of extending the Type A test interval for the CNS to age-related liner corrosion.

The prior assessment included the increase in containment leakage for EPRI Containment Failure Class 3 leakage pathways that are not included in the Type B or Type C tests. These Classes (3a and 3b) include the potential for leakage due to flaws in the containment shell. The impact of increasing the ILRT Interval for these classes included the probability that a flaw would occur and be detected by the Type A test that was based on historical data. Since the historical data includes all known failure events, the resulting risk impact inherently includes that due to age-related degradation.

The present analysis is intended to provide additional assurance that age-related liner corrosion will not change the conclusions of the prior assessment. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP – Reference A1), Comanche Peak Steam Electric Station (CPSES - Reference A2), D. C. Cook Nuclear Plant (CNP – Reference A3) and St. Lucie (SL – Reference A4) in responses to requests for additional information from the NRC staff. The CCNPP, CPSES and CNP extension request submittals have been approved by the NRC.

The significantly lower potential for corrosion of freestanding steel shell containments, such as that at CNS, is considered. This is due to the significantly smaller surface area susceptible to corrosion resulting from foreign material imbedded in concrete contacting the steel containment. Because of this, the analysis is carried out separately for those portions of the containment not in potential contact with foreign material and those portions in potential contact with the foreign material.

As in Reference A1, this calculation uses the following steps with CNS values utilized where appropriate:

Step1 – Determine corrosion-related flaw likelihood.

Historical data will be used to determine the annual rate of corrosion flaws for the containment. The significantly lower potential for corrosion in the CNS containment will be included.

Step 2 – Determine age-adjusted flaw likelihood.

The historical flaw likelihood will be assumed to double every 5 years. The cumulative likelihood of a flaw is then determined as a function of ILRT interval.

Step 3 – Determine the change in flaw likelihood for an increase in inspection interval.

The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests is then determined from the results of Step 2.

Step 4 – Determine the likelihood of a breach in containment given a flaw.

For there to be a significant leak from the containment, the flaw must result in a gross breach of the containment. The likelihood of this occurring is a function of pressure and is evaluated at the CNS ILRT pressure.

Step 5 – Determine the likelihood of failure to detect a flaw by visual inspection.

The likelihood that the visual inspection will fail to detect a flaw will be determined considering the portion of the containment that is uninspectable at CNS as well as an inspection failure probability.

Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood that the increase in test interval will result in a containment leak not detected by visual examination is determined as the product of the increase in flaw

likelihood due to the increased test interval (Step 3), the likelihood of a breach in containment (Step 4) and the visual inspection non-detection likelihood (Step 5). The results of the above for the two regions of the containment are then added to get the total increased likelihood of non-detected containment leakage due to age-related corrosion resulting from the increase in ILRT interval.

The result of Step 6 is then used, along with the results of the prior risk analysis in the body of this analysis to determine the increase in LERF as well as the increase in person-rem/year and conditional containment failure probability due to age-related liner corrosion.

A.4. Input Information

1. General methodology and generic results from the Calvert Cliffs assessment of age-related liner degradation (Reference A1).

2. The CNS ILRT test pressure of 58.0 psig (Reference A5).

3. CNS containment failure pressure of 300 psia based on Drywell rupture at temperatures less than 500 degrees F. (Section 3.4.1 Reference A6).

4. The surface area of the containment potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access is 28, 867 ft^2. As depicted in Figure A-1, this is based on surface area of Drywell Zones I, II, III, & IV plus the LOCA Vent System. (Reference A7). Note that this is conservative since much of the containment surface area does not adjacent to concrete.

5. The number of containments, either free-standing steel shell or concrete with steel liners is 104, and the average area of steel potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access, is 61,900 ft^2 (Reference A11).

A.5. References

A1. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, "Constellation Nuclear letter to USNRC, March 27, 2002.

A2. "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and

50-446, Respond to Request for Additional Information Regarding License Amendment Request (LAR) 01-14 Revision to Technical Specification (TS) 5.5.16 Containment Leakage Rate Testing Program," TXU Energy letter to USNRC, June 12, 2002.

A3. "Donald C. Cook Nuclear Plants Units 1 and 2, Response to Nuclear Regulatory Commission Request for Additional Information Regarding the License Amendment Request for a One-time Extension of Integrated Leakage Rate Test Interval," Indiana Michigan Power Company, November 11. 2002.

A4. "St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Proposed License Amendments, Request for Additional Information Response on Risk-Informed One Time Increases in Integrated Leak Rate Test Surveillance Interval," Florida Power & Light Company letter to USNRC, December 13, 2003.

A5. Cooper Nuclear Station Procedure 6.PC.504, "Primary Containment Integrated Leakage Rate Tests."

A6. Cooper Nuclear Station Level II PSA 1998 Update, November 1998.

A7. GE Specification 22A5742 Revision 4, CNS File # 70019726.

A8. "Containment Liner Through Wall Defect due to Corrosion," Licensing Event Report, Ler-NA2-99-02, North Anna Nuclear Power Plant Station Unit 2.

A9. "Brunswick Steam Electric Plant, Units 1 and 2, Dockets 50-325 and 50-324/License Nos. DPR=71 and DPR-62, Response tp Request for Additional Information Regarding Request for License Amendments – Frequency of Performance Based Leakage Rate Testing," CP&L letter to USNRC, February 5, 2002.

A10. "IE Information Notice No. 86-99; Degradation of Steel Containments." USNRC, December 8. 1986.

A11. E. R. Schmidt, "Calculation of Industry Average Containment Surface Area Subject to Age-Related Corrosion Due to Foreign Material," Analysis File 17547-0001-A4, Rev. 0, November 14, 2003.

A.6. Major Assumptions

1. There has been four instances of age-related corrosion that resulted in holes in steel containment liners or shells. Three were in concrete containments with steel liners and due to foreign material imbedded in the concrete in contact with the steel liner (Cook - Reference A3, North Anna – Reference A8 and Brunswick – Reference A9). The fourth was in a freestanding steel containment and occurred in an area where sand fills the gap between the steel shell and the surrounding concrete and was attributed to water accumulating in this sand

(Oyster Creek – Reference A10).Based on this data, corrosion induced failures are only postulated to occur in the areas of the CNS containment steel shell in contact with concrete or other areas where foreign material may be trapped. For the other areas where the containment steel shell is not likely to be in contact with foreign material, the corrosion induced failure rate should be substantially lower and taken to be that based on no observations of corrosion induced failure of the containment steel shell in these regions.

2. The historical data of age-related corrosion leading to holes in the steel-containment has occurred primarily (3 out of 4 instances) for steel lined concrete containments. For these containments the surface area in contact with the concrete comprises essentially the entire surface area of the containment. As depicted in Figure A-1, it was conservatively assumed that the surface area of Drywell Zones I, II, III, & IV including the LOCA Vent [A7] is in contact with the concrete. This surface area is 28,867 square feet. Since the greater the surface area in contact with the concrete the greater the chance of foreign material being in contact with steel containment, and therefore the greater the chance of corrosion-induced flaws, the containment failure rate due to corrosion is assumed proportional to the surface area in contact with the concrete. The CNS containment failure rate due to corrosion will be that for the industry times the ratio of the surface area at risk for CNS to the average area at risk for the industry.

3. The visual inspection data is conservatively limited to 5.5 years reflecting the time from September 1996, when 10 CFR 50.55a started requiring visual inspection, through March 2002, the cutoff date for this analysis. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to September 1996 (and after March 2002) and there is no evidence that liner corrosion issues were identified. (Step 1)

4. As in Reference A1, the containment flaw likelihood is assumed to double every 5 years. This is included to address the increased likelihood of corrosion due to aging. (Step 2)

5. The likelihood of a significant breach in the containment due to a corrosion induced localized flaw is a function of containment pressure. At low pressures, a breach is very unlikely. Near the nominal failure point, a breach is expected. As in Reference A1, anchor points of 0.1% chance of cracking near the flaw at 20 psia and 100% chance at the failure pressure (Drywell failure pressure of 300 psia for CNS from Reference A6) are assumed with logarithmic interpolation between these two points. (Step 4)

6. In general, the likelihood of a breach in the lower head region of the containment, and this breach leading to a large release to the atmosphere, is less than that for the cylindrical portion of the containment. The assumption discussed in item 5 above is, however, conservatively applied to the lower head region of the containment, as well as to the cylindrical portions.

7. All non-detected containment overpressure leakage events are assumed to be large early releases.

8. The interval between ILRTs at the original frequency of 3 tests in 10 years is taken to be 3

years.

A.7. Identification of Computer Codes

EXCEL Spreadsheets CNS_ILRT_EPRI-104284_r1.xls, CNS_ILRT_NEI-2001_r3.xls, and CNS_ILRT_EPRI-1009325_r1.xls.

A.8. Detailed Analysis

A.8.1 Step 1 – Determine the corrosion-related flaw likelihood.
As discussed in Assumptions 1, 2 and 3, the likelihood of through-wall defects due to corrosion for the areas of the containment potentially contacted by foreign materials is based on 4 data points in 5.5 years.

[4 failures*(28,867ft^2 / 61,900ft^2 / (104 plants*5.5 years/plant) = 3.26E-3 per year]

For the areas of the containment where foreign material is not likely to contact the containment the defect likelihood is taken to be that for no observed failures using a non-informative prior distribution.

Failure Frequency = [# of failures (0) + ½] / (Number of unit years (104*5.5)]
= 8.74E-4 per year.

A similar area-at-risk correction as above for the area in contact with concrete is not appropriate for the area where foreign material is not likely to contact the containment since the majority of the steel liner or shell for all plants has at least one side of the surface subject to this reduced corrosion (and none has been observed).

A.8.2 Step 2 – Determine age-adjusted liner flaw likelihood.

Reference A1 provides the impact of the assumption that the historical flaw likelihood will double every 5 years on the yearly, cumulative and average likelihood that an age-related flaw will occur. For a flaw likelihood of 5.2E-3 per year, the 15 year average flaw likelihood is 6.27E-3 per year for the cylinder/dome region. This result of Reference A1 is generic in nature, as it does not depend on any plant-specific inputs (i.e., except the assumed historical flaw likelihood).

For the present assumption of 4 historical failures in 104 plants, the 15 year average flaw likelihood is 62.7% (3.26E-3/5.2E-3 = 0.627 or 62.7%) of the above value (6.27E-3) or 3.93E-3 per year, and in accordance with Assumption 1, is applicable to the region of the containment potentially in contact with foreign material.

Similarly, for the region of the containment not potentially in contact with foreign material, the 15-year average flaw likelihood is 16.8%(8.74E-4/5.2E-3 = 0.168) of the above value (6.27E-3) or 1.05E-3 per year.

A.8.3 Step 3 – Determine the change in flaw likelihood for an increase in inspection interval. The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is determined from the result of Step 2 in Reference A1 to be 8.7% for the cylinder/dome region based on assumed historical flaw likelihood and the resulting 6.27E-3 per year 15 year average flaw likelihood. This result of Reference A1 is generic in nature, as it does not depend on any plant specific, inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 62.7% (as in Step 2) of that given in Reference A1 (0.627*8.7%) or 5.45% and in accordance with Assumption 1 is applicable to only the region of the containment potentially in contact with foreign material.

Similarly, for the region of the containment not potentially in contact with foreign material, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 16.5% (as in Step 2) of that given in Reference A1 or 1.46%.

A.8.4 Step 4 – Determine the likelihood of a breach in containment given a liner flaw. The likelihood of a breach in containment occurring is determined as a function of pressure as follows:

For a logarithmic interpolation on likelihood of breach

$$\text{LOG (likelihood of breach)} = m\ (\text{pressure}) + a$$

Where m = slope
a = intercept

The values of m and a are determined from solution of the two equations for the values of 0.1% at 20 psia and 100% of containment failure pressure at 300 psia (Reference A6).

$$\text{Log } 0.1 = m*20 + a$$

$$\text{Log } 100 = m*300 + a$$

Solving for "m",

$$m = (\text{Log } 100 - \text{Log } 0.1)/(300-20) = 0.01071$$

Solving for "a",

$$a = \text{Log } 0.1 - 0.01071*20 = -1.2143$$

The upper end of the range of CNS ILRT pressure of 58.0 psig (Reference A5) gives the highest likelihood of breach.

At 72.7 psia (58+14.7) the above equation gives

Log (Likelihood of breach) = 0.01071*72.7-1.2143 = -0.435667

Likelihood of breach $=10^{-0.45679} = 0.37\%$

In accordance with Reference A1, the above value is for the cylinder/dome portions of the containment. For this analysis, this value is also assumed to be applicable to the region of the containment potentially in contact with foreign material.

A.8.5 Step 5 - Determine the likelihood of failure to detect a flaw by visual inspection

A review of the geometry of the containment shell and the relative areas that are not inspectable and those in potential contact with foreign material, indicates that these two areas are essentially the same, both comprising approximately 50% of the total surface area of the steel shell (Reference A7). Consequently, the portion of the containment not likely to be in contact with potential foreign material is 100% visually inspectable, while the portion that may be in contact with potential foreign material is not visually inspectable. A 10% failure rate for that portion of the containment that is visually inspectable is assumed.

A.8.6 Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood of non-detected containment leakage in each region due to age-related corrosion of the liner considering the increase in ILRT interval is then given by:

The increased likelihood of an undetected flaw because of the increased ILRT interval (Step3)	*	The likelihood of a containment breach given a liner flaw (Step 4)	*	The likelihood that visual inspection will not detect the flaw (Step 5)

1.46% * 0.0037*0.10 = 0.0005% for the regions not potentially contacted by foreign material

5.46% * 0.0037*1.0 = 0.0202% for the regions potentially contacted by foreign material

The total is then the sum of the values for the two regions or

Total likelihood of Non-Detected Containment Leakage = 0.0005% + 0.0202% = 0.0207% for ILRT interval increase from 3 years to 15 years.

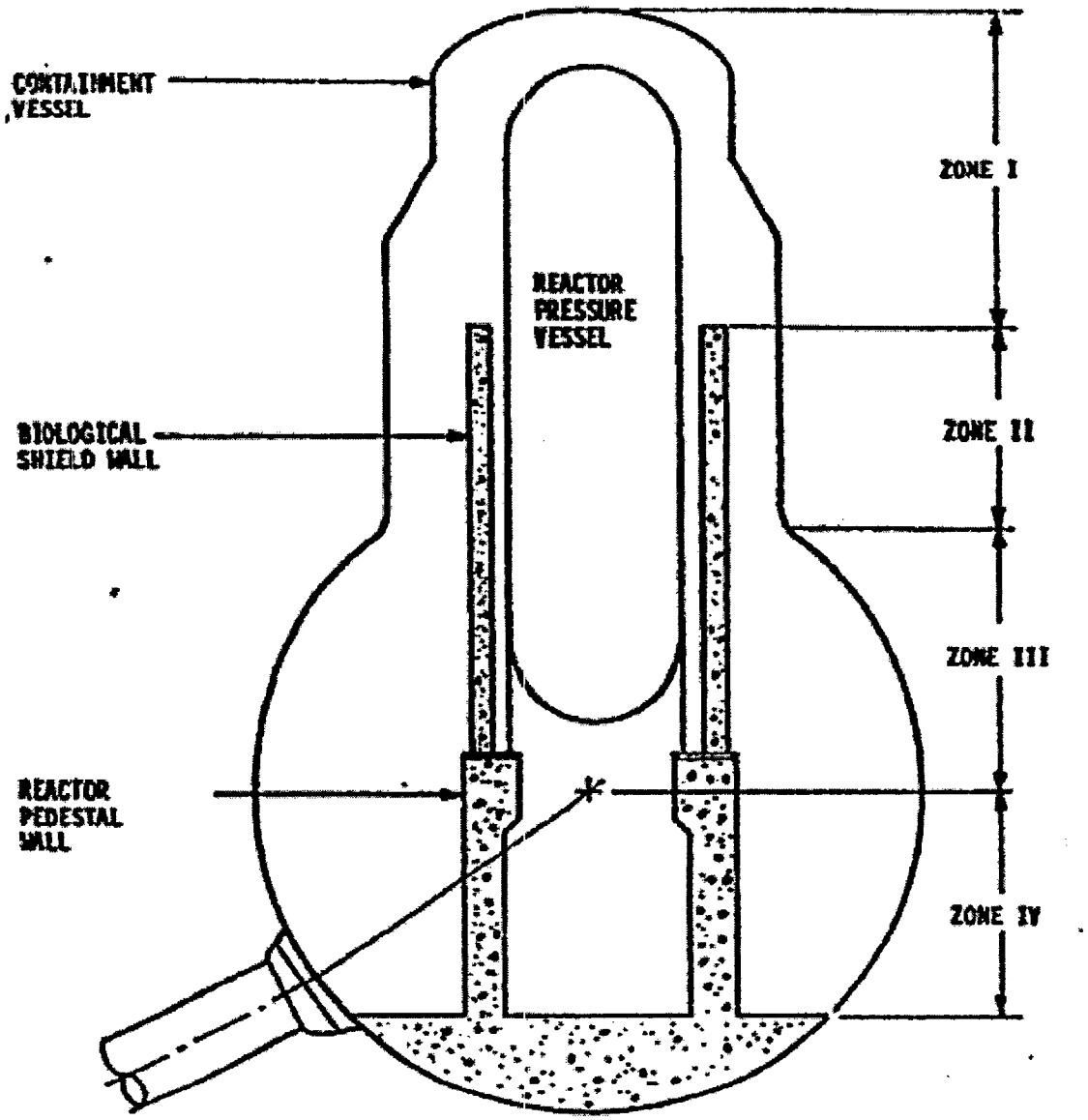


Figure A-1: Definition of CNS Containment Zones

APPENDIX B – EXTERNAL EVENT IMPACT

EFFECT OF EXTERNAL EVENTS ON RISK INFORMED/RISK IMPACT ASSESSMENT FOR EXTENDING CONTAINMENT TYPE A TEST INTERVAL

This appendix discusses the external events assessment in support of the CNS ILRT interval extension risk assessment. External hazards were evaluated in the CNS Individual Plant Examination of External Events (IPEEE) Submittal [B2] in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4).The IPEEE Program was a one-time review of External hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Although the external event hazards in the CNS IPEEE were evaluated to varying levels of conservatism, the results of the CNS IPEEE are nonetheless used in this risk assessment as a sensitivity study to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

B.1. High Winds, Flooding. Transportation, and Nearby Industrial Facility Accidents

The IPEEE assessment [B2] determined that the risk due to high winds, external flooding, transportation, and nearby industrial facility accidents which might lead to core damage were below the screening criteria frequency of less than 1.0E-6/yr.Therefore, these external events are not evaluated further in this calculation and are expected to have an insignificant impact on the results of this calculation.

B.2. Fire

The IPEEE assessment [B2] utilized the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. The results of the CNS IPEEE showed that postulated fire events at CNS does not constitute a significant contributor to overall core damage risk The majority of the fire areas had calculated core damage frequencies less than 1.0E-6/yr. Table B-1 displays the fire areas that had calculated screening values greater than the FIVE screening criteria of 1.0E-6/yr. However, because of the conservatism included in the FIVE screening analysis, it is judged that the calculated values should be lower. No induced containment performance vulnerabilities were for these areas were identified. Therefore, fires do not result in or cause containment breach concerns beyond those already addressed in the CNS risk model.

Table B-1: Core Damage Frequency Estimates for Unscreened Fire Compartments

FIRE COMPARTMENT	CORE DAMAGE FREQUENCY (PER-YR)
Division I Essential Switchgear 1F Room (Compartment 3A)	1.11E-6
Division II Essential Switchgear 1G Room (Compartment 3B)	2.72E-6
Control Room and SAS Corridor (Compartment 10B)	3.73E-6
Service Water Pump Room (Compartment 20A)	1.33E-6
CDF_{FIRE}	8.89E-6

This information is used in Section B.4 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment

B.3. Seismic

The CNS IPEEE assessment [B2] documented completion of a focused scope Seismic Margins Assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic process which does not calculate risk values.

Reference [B1] provides a simplified methodology (Simple Hybrid Method) for estimating the seismic risk based on a SMA analysis. It has shown that only the individual plant High Confidence Low Probability of Failure (HCLPF) seismic capacity is required in order to estimate the seismic CDF within an approximate factor of two. This approach has been used in previous NRC submittals. The approach is:

Step 1: Determine the CNS HCLPF seismic capacity (C_{HCLPH}) from the SMA analysis

Step 2: Estimate the 10% conditional failure probability capacity ($C_{10\%}$) from

$$C_{10\%} = F_\beta * C_{HCLPH}$$

$$F_\beta = e^{1.044\beta}$$

Where1.044 is the difference between the 10% NEP standard normal variable (-1.282) and the 1% NEP standard normal variable (-2326).

Experience gained from previous high quality seismic PRA's indicates the plant damage state fragility determined by rigorous convolution will tend to have β_c values in the range of 0.30 to 0.35. Therefore, the Simplified Hybrid Model recommends:

$$C_{10\%} = 1.4 * C_{HCLPH}$$

Step 3: Determine the hazard exceedance frequency ($H_{10\%}$) that corresponds to $C_{10\%}$ from the hazard curves.

Step 4: Determine the seismic risk $CDF_{SEISMIC}$ (i.e., seismic related CDF) from:

$$CDF_{SEISMIC} = 0.5 * H_{10\%}$$

Using the above steps the Simplified Hybrid Model can be applied to CNS to estimate the seismic risk below:

Step 1: The plant HCLPF is determined in References [B4] and [B5] to be at least 0.3g peak ground acceleration (PGA).

Step, 2: Using the relationship described above:

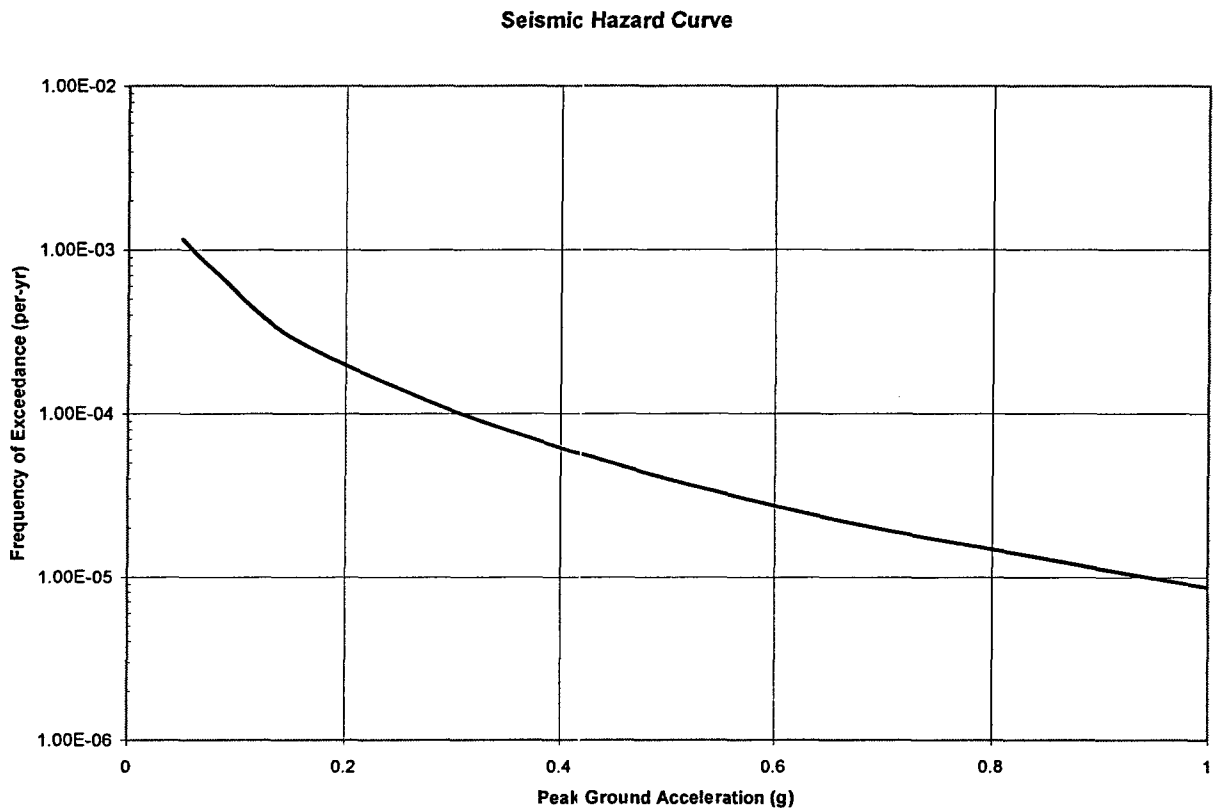
$$C_{10\%} = 1.4 * 0.3g \text{ PGA} = 0.42g \text{ PGA}$$

Step 3: The seismic hazard curve for CNS is obtained from Reference [B3]. It is replicated below with the CNS HCLPF of 0.42g PGA estimated from the available data points and added to Table B-2.

Table B-2: CNS Seismic Hazard Curve (From NUREG-1488)

Peak Ground Acceleration (g)	Mean Annual Exceedance Frequency
0.05	1.155E-03
0.08	7.383E-04
0.15	2.924E-04
0.26	1.335E-04
0.31	9.828E-05
0.41	5.867E-05
0.42[1]	5.662E-05[1]
0.51	3.813E-05
0.66	2.211E-05
0.82	1.392E-05
1.02	8.187E-06

NOTE (1): The value of 5.662E-5/yr for 0.42g was obtained from a best fit curve based on points on the seismic hazard curve.



Step 4: Using the recommended relationship described above:

$$CDF_{SEISMIC} = 0.5 * H_{10\%} = 0.5 * 5.662E\text{-}5/yr = 2.831E\text{-}5/yr$$

This information is used in Section B.4 of this appendix to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

B.4. Impact of External Events on LERF

Based on the previous discussion in Sections B.1 through B.3, the total CNS external event initiated CDF is approximately 8.89E-6/yr (internal fires) + 2.831E-5/yr (seismic) =3.72E-5/yr.

For seismic risk, the Simplified Hybrid Model provides an overall estimate of seismic risk, but does not provide information as to the specific accident sequences. Similarly for the fire risk, the FIVE results do not easily provide information as to the specific accident sequences. Conservatively, the full estimated seismic CDF and full estimated fire CDF was used to calculate the 3b frequency and were not adjusted for sequences that independently cause LERF or will

never cause LERF. In order to assess the impact of external events on the proposed ILRT extension request, the impact on LERF was assessed in accordance with the NEI Interim Guidance described in Section 9.0. The NEI Interim Guidance was used because it yields the most conservative results relative to the other two approaches used in this calculation.

The impact on 3b frequency due to increases in the ILRT surveillance interval was calculated for external events using the relationships described in Section 9.0. The EPRI Category 3b frequencies for the10-year and 15-year ILRT intervals were quantified using the total external event CDF. The change in the LERF risk measure due to extending the ILRT interval from 10 years to 15 years, including both internal and external hazard risk, is provided on Table B-3.

Table B-3: Calculation of LERF Impact Including External Events using NEI Interim Guidance

	3b Frequency (1-per-10 year ILRT)	**3b Frequency (1-per-15 year ILRT)**	**LERF Increase**
External Events Contribution	3.35E-07	5.02E-07	1.67E-07
Internal Events Contribution	5.27E-09	7.90E-09	2.63E-09
Combined (Internal + External)	3.40E-07	5.10E-07	1.70E-07

Thus, the estimated increase in LERF due to external events is 1.67E-7/yr. When the internal and external events contributions are summed, the total estimated increase in LERF is 1.70E-7/yr. This increase is in the range of 1E-07/yr to 1E-06/yr, which is in Region II of the RG 1.174 LERF acceptability curve. However, based on the conservative nature of this sensitivity study, it is expected that a more detailed external event study would reduce the estimated increase in LERF to less than 1E-07/yr. For example, this study counted the full estimated seismic CDF and full estimated fire CDF against the 3b frequency. Table 8-6 shows that the 3b frequency is 0.11% of the total Internal Events CDF for the 15 year ILRT Test Interval. It should be noted that Reference [B2] did not identify any unique containment vulnerabilities for the seismic and fire evaluations.

The acceptance guidelines for Region II of the RG 1.174 LERF acceptability curve are that it can be reasonably shown that the total LERF is less than 1E-05/yr and that the cumulative changes be tracked. The baseline LERF is 5.60E-07. Based the LERF increase calculated using the NEI Interim Guidance (i.e., 1.70E-07), the total LERF for the requested change is 7.30E-07/yr which meets the total LERF criterion.

B.5. References

B1. Reference: R. P. Kennedy, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August, 1999.

B2. NLS960134, CNS Letter from G. R. Horn to U. S. Nuclear Regulatory Commission, "Individual Plant Examination for External Events (IPEEE) Report -10 CFR 50.54(f) Cooper Nuclear Station NRC Docket No. 50-298, License No. DPR-46," October 30, 1996

B3. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Plant Sites East of the Rocky Mountains," October 1993.

B4. NLS99008 CNS Letter from John H. Swailes to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information – Individual Plant Examination for External Events (IPEEE) Cooper Nuclear Station, NRC Docket 50-298, DPR-46," January 28, 1999.

B5. Letter from United States Nuclear Regulator Commission, "Cooper Nuclear Station – Review of Individual Plant Examination of External Events (TAC No. 83611)," April 27, 2001.