Entergy

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PNP 2011-018

April 6, 2011

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

SUBJECT: License Amendment Request to Extend the Containment Type A Leak Rate Test Frequency to 15 Years

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

REFERENCE: 1. Letter from NRC to Nine Mile Point Nuclear Station, LLC, dated March 30, 2010, Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650) (ADAMS Accession Number ML100730032)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PLP). ENO proposes to revise Appendix A, Technical Specifications (TS), to allow extension of the ten-year plus 15-month frequency of the PLP Type A or Integrated Leak Rate Test (ILRT) that is required by Technical Specification (TS) 5.5.14 to 15 years on a permanent basis.

A similar LAR to extend the ILRT interval to 15 years was approved for the Nine Mile Point Unit No. 2 (Reference 1)

This proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. The bases for this determination are included in Attachment 1, which provides a description of the proposed change, background discussion, technical analysis, regulatory analysis, and environmental review. Attachment 2 provides the revised TS pages reflecting the proposed changes. Attachment 3 provides the annotated TS pages showing the proposed changes.

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Attachment 4 provides an assessment on the risk impact of extending the ILRT interval. Attachment 5 provides a commitment list.

ENO requests approval of the proposed amendment by April 7, 2012. Once approved, the amendment will be implemented within 30 days.

In accordance with 10 CFR 50.91, ENO is notifying the State of Michigan of this proposed license amendment by transmitting a copy of this letter to the designated state official.

Summary of Commitments

The proposed change includes one new commitment. There are no revisions to existing commitments. The new commitment is summarized in Attachment 5.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 6, 2011.

Sincerely,

Thomas Kirwin

tpk/jlk

Attachments:
1. Description and Evaluation of Requested Change
2. Renewed Operating License Page Change Instructions and Revised Technical Specifications Page
3. Mark-up of Technical Specifications Page
4. Risk Impact Assessment of Extending Palisades ILRT Interval
5. List of Regulatory Commitments

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

ATTACHMENT 1
DESCRIPTION AND EVALUATION OF REQUESTED CHANGE

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PLP). ENO proposes to revise Appendix A, Technical Specifications (TS), to allow extension of the ten-year frequency of the PLP Type A or Integrated Leak Rate Test (ILRT) that is required by Technical Specification (TS) 5.5.14 to 15 years on a permanent basis.

The proposed amendment would revise PLP TS 5.5.14, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2-A, dated October 2008, as the implementation document used by ENO to develop the PLP performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J (Option B).

Revision 2-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163. In the safety evaluation (SE) issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

In accordance with the guidance in NEI 94-01, Revision 2-A, ENO proposes to extend the interval for the primary containment ILRT, which is currently required to be performed at ten year intervals to no longer than 15 years from the last ILRT. The next ILRT is currently due no later than August 3, 2012, as required by TS 5.5.14a. This is approximately 11.25 years since the last ILRT was completed on May 3, 2001. A one-time 15-month extension to the ten-year ILRT frequency was approved, by the NRC, in license amendment no. 240, issued August 23, 2010. With the current 15-month extension, the next ILRT would be performed during the spring 2012 refueling outage. The proposed amendment would allow the next ILRT for PLP to be performed within 15 years from the last ILRT (i.e., by May 3, 2016), as opposed to the current interval. The change would allow successive ILRTs to be performed at 15-year intervals (assuming acceptable performance history). The performance of fewer ILRTs would result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

2.0 PROPOSED CHANGE

The PLP containment leakage rate testing program TS 5.5.14a. 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 of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the next Type A test performed after the May 3, 2001, Type A test shall be performed no later than August 3, 2012, as modified by the following exceptions:"

The proposed change would revise TS 5.5.14a., by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A. Additionally, the information related to the last Type A test performance date and the next required Type A test date would be deleted from the TS. Removal of the Type A test date information from the TS would remove out-of-date information and remove an unnecessary administrative burden for ENO and the NRC. If the date information were to be retained and modified with the next upcoming Type A test due date, by May 3, 2016, it would result in a future administrative burden to remove the out-of-date information. The revised TS 5.5.14a. would read as follows:

> "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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:"

3.0 BACKGROUND

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

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

Also in 1995, RG 1.163 was issued. The RG endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency from the containment Type A (ILRT) test from three tests in ten years to one test in ten years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, "Performance-Based Containment Leak-Test Program," and Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," both of which illustrated that the risk increase associated with extending the ILRT surveillance interval was very small.

By letter dated December 7, 2000, as revised by letter dated January 12, 2001, Consumers Energy (the former PLP owner and licensee) submitted a TS change request concerning the implementation of 10 CFR 50, Appendix J, Option B. In the SE approving this request (amendment no. 194 issued in NRC letter of March 30, 2001), the NRC noted the proposed TS changes were in compliance with the requirements of Option B, and are consistent with the guidance in RG 1.163. With the approval of the amendment, PLP transitioned to a performance-based ten year frequency for the Type A tests.

ENO submitted a LAR to extend the ILRT interval from ten years (120 months) to approximately 135 months via letter dated August 25, 2009, and supplemented by letter dated May 3, 2010. This one-time extension was approved by the NRC, as license amendment no. 240, in letter dated August 23, 2010.

By letter dated August 31, 2007, NEI submitted NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC Staff for review.

NEI 94-01, Revision 2, describes an approach for implementing the optional performance-based requirements of Option B, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals. However, the NEI guideline does not address how to perform the tests because these details are included in other industry documents (e.g., American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002). The NRC final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 has subsequently been issued as Revision 2-A dated October 2008.

EPRI Report No. 1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to 15 years, using current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The NRC's final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of EPRI Report No. 1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI Report No. 1009325 has subsequently been issued as Revision 2-A (also identified as Technical Report TR-1018243) dated October 2008.

4.0 TECHNICAL ANALYSIS

As required by 10 CFR 50.54(o), the PLP containment is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the PLP 10 CFR 50 Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0. This LAR proposes to revise the PLP 10 CFR 50, Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 2-A.

In the June 25, 2008 NRC SE, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SE. The following Table 4.0, SE Section 4.1 Limitations and Conditions Response, addresses each of the six limitations and conditions.

Table 4.0
SE Section 4.1 Limitations and Conditions Response

Limitations and Conditions (from Section 4.1 of SE)	ENO Response
1. For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002.	Following the NRC approval of this LAR, ENO would use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future PLP Type A tests are performed (see Attachment 5, "List of Regulatory Commitments").
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	A schedule of containment inspections is provided in Section 4.3 below.
3. The licensee addresses the areas of the containment structure potentially subjected to degradation.	General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is typically conducted in accordance with the PLP Containment Inservice Inspection Plan which implements the requirements of the ASME, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). There are no primary containment surface areas that require augmented examinations in accordance with ASME Section XI, IWE-1240.
4. The licensee addresses any test and inspections performed following major modifications to the containment structure, as applicable.	The Entergy fleet design change process would address any testing and inspection requirements following future major modifications to the containment structure. This process provides a disciplined approach for determining the program and system interfaces associated with design changes. Specific questions are provided in this process pertaining to the ASME Containment In-Service Inspection Program, ASME Appendix J (Primary Containment Leak Rate Testing) Program, and ASME Section XI

	Repair/Replacement Program. These questions prompt the responsible engineer to consult with the applicable program owner for required actions including testing and inspections.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.	ENO acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27 dated December 8, 2008.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. PLP is not licensed pursuant to 10 CFR Part 52.

To comply with the requirement of 10 CFR 50, Appendix J, Option B, Section V.B, PLP TS 5.5.14a. currently references RG 1.163. RG 1.163 states that NEI 94-01, Revision 0, provides methods acceptable to the NRC for complying with Option B, with three exceptions described therein.

The three exceptions to the guidelines of RG 1.163, in TS 5.5.14a. are as follow:

1. Leakage rate testing is not necessary after opening the Emergency Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at $\geq$10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

These current TS 5.5.14a. exceptions to the RG 1.163 guidelines would be maintained as part of the revised PLP TS.

No modifications that require a Type A test are planned prior to the 2015 refueling outage,1R24, when the next Type A test would be performed under this proposed change. Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. Additionally, there have been no pressure or temperature excursions in the containment, which could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within containment, which could affect leak-tightness.

4.1 <u>Previous ILRT Results</u>

Previous ILRT testing confirmed that the PLP containment structure leakage is acceptable with respect to the TS acceptance criterion of 0.1% of containment air weight at the design basis loss of coolant accident pressure (L_a). Since the last two PLP Type A as-found results were less than 1.0 L_a, a test frequency of at least once per 15 years would be in accordance with NEI 94-01, Revision 2-A.

The first PLP ILRT was completed on May 26, 1970. Subsequent PLP ILRTs were completed on May 2, 1974, March 28, 1978, November 18, 1981, January 25, 1986, November 5, 1988, February 17, 1991, and May 3, 2001. The second, third, and fourth post-operational tests, which were completed in March 1978, November 1981, and January 1986 respectively, resulted in the combined calculated leakage plus the adjusted measured penetration leakage exceeding the acceptance criteria. A Final Safety Analysis Report (FSAR) historical summary of these three tests is provided later, beginning on page 9. There have been no other failed ILRTs at PLP.

The following as-found and as-left Type A post-operational test results provide comparison to the allowable leakage rates specified in the PLP TS at the time the Type A tests were performed.

<u>April 30, 1974</u> – The calculated leak rate at reduced test pressure (28 psig) was 0.0342 wt%/day of contained air. The maximum allowable leak rate (L_t) at this pressure was 0.0514 wt%/day per the then current TS. In accordance with the then existing version of Appendix J, the allowable operational leak rate was 75% of the maximum allowable in order to allow for possible degradation. Therefore, the allowable operational leak rate per Appendix J of 0.0386 wt%/day was met.

<u>March 28, 1978</u> – As-found leakage rate at 28 psig of -0.00708 wt %/day with a 95% upper confidence limit of 0.00195 wt %/day and a TS limit of 0.0559 wt %/day. The negative leakage rate was attributed to instrument error. During the test, a containment leak was discovered on the containment purge air exhaust penetration fitting. This

fitting was replaced. The leak rate from this fitting was added to the test resultant leak rate of -0.00708 wt %/day, which provided an as-found leakage of 0.09242 wt %/day.

November 18, 1981 – The calculated nominal leakage rate was 0.0326 wt%/day, with a one sided 95% upper confidence limit of 0.0349 wt%/day. Assuming the containment pressure of 28 psig, the maximum allowable leakage rate was 0.0713 wt%/day. The required leakage rate could not exceed 75% of the allowable leakage rate or 0.0535 wt%/day.

January 1986 – A reduced test pressure (P_t) of 28.25 psig was recorded at the end of the 30-hour hold test. At the reduced test pressure (P_t), the measured containment leak rate (L_{tm}) for the hold test was 0.0157 wt%/day with a 95% upper confidence limit of 0.0187 wt%/day. After upward adjustment to accident pressure (Pa), the measured leak rate (L_{am}) was 0.0262 wt%/day at the 95% upper confidence limit. The calculated as-left total containment leak rate was 0.0290 wt%/day at the 95% upper confidence limit, and included Type C test additions for systems not in their accident status during the conduct of the Type A test and compensating for a 2% increase in the pressurizer level (equivalent to 0.0017 wt%/day).

Per 10 CFR 50 Appendix J, Section III.A.5.b.1, the maximum acceptable leak rate for a Type A test is 0.075 wt%/day (0.75 L_a), at the PLP accident pressure (P_a) of 55 psig. This includes corrective additions to account for omitted systems and containment free volume changes.

Following penalty additions for outage repairs resulting in improvements in Type B/C leak rates, the as-found Type A leak rate was 0.1061 wt%/day at the 95% upper confidence limit, which did not meet the acceptance criteria of 0.075 wt%/day (0.75 L_a). Two-thirds of the total penalty assessed was based on a conservative minimum pathway determination for penetration numbers 40 and 69 necessitated through replacement and/or repair of double isolation valves.

November 1988 – The measured containment leak rate (L_{tm}), at the reduced test pressure of 28.66 psig, was 0.01651 wt%/day, with a 95% upper confidence limit of 0.01758 wt%/day. Adjustment of the reduced pressure leak rate upward to accident pressure (P_a) resulted in a measured leak rate (L_{am}) of 0.0231 wt%/day with a 95% confidence limit of 0.0246 wt%/day. The addition of Type B & C penalties including compensation, for a one percent increase in pressurizer level, resulted in a calculated as-left total containment leak rate of 0.02617 wt%/day at the 95% upper confidence limit adjusted to P_a.

The as-found Type A leak rate was 0.02915 wt%/day at the 95% upper confidence limit or 0.0408 wt%/day (95% upper confidence limit) corrected upward for accident pressure (P_a) of 55 psig. This value was within the acceptance criteria of 0.075 wt%/day (0.75 L_a).

February 1991 – The measured containment leak rate (L_{am}) for the hold test at 55.61 psig was 0.02473 wt%/day with an upper 95% confidence level of 0.0700838 wt%/day. The addition of the Type B and C penalties resulted in a calculated as-left total containment leak rate of 0.070439 wt%/day. This leak rate was within the 10 CFR 50, Appendix J, Section III.A.5.b.1, maximum acceptable leak rate for a Type A test at the accident pressure (P_a) of 55 psig of 0.075 wt%/day (0.75 L_a), including the corrective additions to account for omitted systems and the containment free volume changes.

The as-found condition is the condition of the containment at the beginning of the outage prior to any repairs or adjustments to the containment boundary. This is normally calculated by reviewing the summary of the local leak rate tests (LLRTs) and calculating the amount of leakage rate improvements due to repairs or adjustments using minimum pathway methodology. This assumes that no major changes to the containment structure were made, but that all leakage improvements were due to penetration repairs or adjustments. However, during the 1990-1991 outage, a construction opening was cut through the containment wall in order to allow replacement of the steam generators. Thus, no correlation could be established between the pre- and post- modification leakage rates. Therefore, the containment ILRT was considered to be a pre-operational test to show that the repairs to the containment adequately met the TS leakage requirements, rather than the performance of a periodic containment ILRT.

<u>May 2001</u> – The pressure across the containment boundary was 53.524 psig (the outside barometric pressure was 14.558 psia). An acceptable mass point leakage rate at the 95% upper confidence limit of 0.0100 wt%/day was determined not including the leakage rate corrections. Total leakage rate corrections (Type B & C LLRT penalties and water volume corrections) were determined to be 0.0022 wt%/day. The as-left 95% upper confidence limit leakage rate was determined to be 0.0122 wt%/day. The maximum allowable leakage rate (L_a) for the containment was 0.1 wt%/day with a test acceptance of 0.075 wt%/day (0.75L_a). The acceptance criteria was met.

The as-found 95% upper confidence limit leakage rate was 0.0140 wt%/day, which included a leakage savings of 0.0018 wt%/day.

<u>FSAR Section 5.8.8.1.1, Historical Summary</u>

The following excerpt from the PLP FSAR describes the reasons for exceeding the acceptance criteria during the second, third, and fourth post-operational ILRT tests. Also provided is information from the NRC Inspection Report dated April 11, 1986, that identified a violation with respect to the methodology used for incorporation of repairs and adjustment leakage rates into the Type A Test results.

> "The Second Postoperational Type A Test - A second postoperational Type A test was completed on March 28, 1978. This was a reduced pressure test, (28 psig). This test was conducted within the general guidelines of the Technical Specifications, 10 CFR 50 Appendix J, and ANSI N45.4.
>
> On February 11, 1975, the Technical Specifications were amended [Amendment 12] and the temperature correction factor was eliminated. The basis for this change was that such a correction is not required by 10 CFR 50 Appendix J, which was

issued several years after the original Technical Specifications. This correction factor was not applied to this test.

During containment pressurization, a leak was found on the 48-inch containment purge air exhaust penetration. The leak was measured and recorded and a grease fitting was replaced to correct the problem. The penetration was then leak tested again. The containment leakage rate after repair of the penetration was well within the acceptance criteria. In fact the calculations showed a net inflow into the containment.

The combined calculated containment leak plus the adjusted measured penetration leak were above the acceptance criteria.

The negative leakage, (net inflow) was attributed to instrument error, which, for this test, was found to be approximately 40% of the acceptance criteria value. Since this magnitude of error can significantly impact future test results, Consumers Power Company [CPCo the former PLP owner] performed a review of test monitoring equipment and procedures. As a result of this review, new instrumentation requirements were established for the ILRT, using ANS 56.8-1981, as a guide.

The Third Postoperational Type A Test - The third postoperational Type A test was concluded on November 18, 1981. This was a reduced pressure test, (28 psig). This test was conducted within the general guidelines of the Technical Specification, 10 CFR 50 Appendix J, and ANSI N45.4.

The containment leakage rate was within the acceptance criteria, but NRC Violation 255\86-005-04 later resulted in a requirement that repairs and adjustments be added to the leak rate. This resulted in the as-found leakage rate exceeding the acceptance criteria.

The Fourth Postoperational Type A Test - The fourth postoperational Type A test was conducted in January 1986. This was a reduced pressure test, (28 psig). This test was conducted within the general guidelines of the Technical Specifications, 10 CFR 50 Appendix J, and ANSI N45.4.

The as-found leakage rate exceeded the acceptance criteria when the repairs and adjustments were added to the calculated leak rate."

NRC Inspection Report dated April 11, 1986, from NRC to CPCo provided the results of a routine inspection that was conducted between January 19 and March 13, 1986. A violation (Violation 255/86005-04(DRS)) was issued with respect to the methodology used at PLP for incorporation of repairs and adjustment leakage rates into the Type A test results. Specifically, it was noted that for an as-found Type A test, CPCo had been performing containment isolation valve leak testing and isolation repairs prior to all of the previous periodic Type A tests without determining the as-found condition of the containment structure. Failure to realize the requirements for an as-found Type A test resulted in PLP failing both the 1978 and 1981 Type A test in the as-found condition.

During this same inspection, it was determined that the 1986 containment ILRT also failed its as-found leakage rate criteria due to the addition of repairs and adjustment penalties. The excessive as-found condition was largely due to the leakage through penetration number 40, reactor coolant sample line, and the uncertainties involved in its local leak rate test. In a letter dated May 9, 1986, from CPCo to NRC, the corrective actions taken to correct the methodology used for the performance of the PLP ILRT were identified. Specifically, the ILRT procedure used at PLP was revised to address adding local leakage differences resulting from repairs.

4.2 Type B and Type C Testing Program

The PLP Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and TS 5.5.14. The Type B and Type C testing program consists of local leak rate testing of penetrations with a resilient seal, double gasketed manways, hatches and flanges, and containment isolation valves that serve as a barrier to the release of the post-accident containment atmosphere.

The piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the containment wall, thus precluding any requirement for expansion bellows.

The last Type A test was performed in 2001. The minimum pathway combined Type B and Type C total leakage value from the 2001 refueling outage is provided below. The data is also provided in terms of percentage of leakage allowed ($0.6L_a$).

The subsequent combined as-found Type B and Type C test values since the 2001 refueling outage are provided in Table 4.2-1 below. The data is also provided in terms of percentage of leakage allowed ($0.6L_a$). The L_a value has not changed during the period since the last Type A test was performed.

Table 4.2-1
Combined As-found Type B & C Test Leakage

Date	As-Found Leakage (sccm)	L_a (sccm)	Percentage (= (As-Found/La)x100)	Percent .6La (As-Found/.6La x 100)
12/27/2001	33,077.7*	148,465	22.28	37.13
4/15/2002	12,989.9	148,465	8.75	14.58
2/6/2003	13,010.8	148,465	8.76	14.61
4/16/2003	14,118.7	148,465	9.51	15.85
12/16/2003	14,488.4	148,465	9.76	16.26
11/11/2004	47,840**	148,465	32.22	53.71
11/15/2004	14928.7	148,465	10.06	16.76
5/4/2006	14,074.7	148,465	9.48	15.80
5/15/2006	15,156.3	148,465	10.21	17.01
1/4/2007	14,840	148,465	10.0	16.66
10/13/2007	16,444.8	148,465	11.08	18.46
11/17/2008	14,669.3	148,465	9.88	16.46
4/18/2009	15,768.72	148,465	10.62	17.70
4/27/2009	15,466.15	148,465	10.42	17.36
7/1/2010	15,466.15	148,465	10.42	17.36
9/13/2010	15,781.05	148,465	10.63	17.72
10/26/2010	18,409.65	148,465	12.40	20.67

* The largest contributor to the as-found leak rate was penetration MZ-66, ILRT Instrument Line, which had a leak rate of 22,142 cc/min. Subsequent to the initial test the penetration was depressurized and retested. The as-left leak rate for MZ-66 was 2,010.5 cc/min, which was acceptable.

** Control Valves CV-1044 and CV-1045 in penetration MZ-69, Clean Waste Receiver Tank Pump Suction, exceeded their administrative limit and were placed on a 30-month test frequency in accordance with the LLRT Program.

Table 4.2-2 (at the end of Attachment 1) provides a description of the Type B and C containment penetrations, test frequencies, dates, and the latest as-left leakage data. Table 4.2-2 also provides information on penetrations that have failed their administrative leakage acceptance criteria.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority (greater than 95%) of all potential containment leakage paths. This amendment request would adopt the guidance in NEI 94-01, Revision 2-A, in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

4.3 <u>Supplemental Inspection Requirements</u>

Prior to initiating a Type A test, a general visual examination of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structure leakage integrity or the performance of the Type A test is performed. This inspection would be conducted in accordance with the PLP Containment Inservice Inspection (ISI) Plan, which implements the requirements of ASME, Section XI, Subsection IWE/IWL. The applicable code edition and addenda for the second ten-year interval IWE/IWL program is the 2004 Edition. There are no relief requests associated with this interval.

The examination performed in accordance with the IWE/IWL program satisfies the general visual examination requirements specified in Option B. Identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A) and (E). Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation are performed in accordance with the requirements of 10 CFR 50.55a(b)(ix)(G) and 10 CFR 50.55a(b)(ix)(H). Each ten-year ISI interval is divided into three approximately equal-duration inspection periods. A minimum of one inspection during each inspection period of the ISI interval is required by the IWE/IWL program. The moisture barrier, as part of the IWE/IWL program will be inspected each period during this ten-year interval. Since a 15-year ILRT interval spans at least four ISI periods, the frequency of the examinations performed in accordance with the IWE/IWL program satisfies the requirement of NEI 94-01, Revision 2-A, Section 9.2.3.2, to perform the general visual examinations during at least three other outages before the next Type A test, if the Type A test interval is to be extended to 15 years. Table 4.3-1 illustrates the inspection periods for the PLP first and second IWE/IWL ISI intervals.

Table 4.3-1
PLP Containment Inservice Inspection Periods (IWE/IWL)

Inspection Interval	Inspection Period	Period Start Date	Period End Date	Refuel Outage	Refuel Year
1	1	October 15, 1999	February 15, 2003	1R14 1R15	1999 2001
1	2	February 16, 2003	June 16, 2006	1R16 1R17 1R18	2003 2004 2006
1	3	June 17, 2006	October 15, 2009	1R19 1R20	2007 2009
2	1	October 16, 2009	February 16, 2013	1R21 1R22	2010 2012
2	2	February 17, 2013	June 17, 2016	1R23 1R24	2013 2015
2	3	June 18, 2016	October 16, 2019	1R25 1R26 1R27	2016 2018 2019

The last PLP Type A test was completed in May 2001 during refueling outage 1R15. Based on a 15-year Type A test interval, the next PLP Type A test would be scheduled for refueling outage 1R24, in 2015 (during Inspection Interval 2, Period 2). Thus, based on the schedule provided in Table 4.3-1 above, three containment general visual examinations performed in accordance with the IWE ISI Program would take place prior to the 2015 Type A test (i.e., during Inspection Interval 1, Periods 2 and 3, and during Inspection Interval 2, Period 1).

Although the inspection periods shown in Table 4.3-1 are based on Subsection IWE inspection requirements, the IWL inspections are typically scheduled in two of the three inspection periods of a 10-year ISI interval, as shown in Table 4.3-1 above. The visual examinations of accessible concrete containment surfaces in accordance with ASME Section XI, Subsection IWL, are performed every five years, resulting in at least two IWL examinations being performed during a 15-year Type A interval.

There are no primary containment surface areas that require augmented examination in accordance with ASME Section XI, IWE-1240.

The following information provides the PLP IWE examination results of the containment metal liner completed during refueling outage 1R18 (2006) and 1R20 (2009). The next IWE examination is scheduled for 1R22 (2012). Also, provided are the PLP IWL examination results of the containment concrete visual inspections completed in 2000, 2005, and 2010 and tendon inspections completed in 2002 and 2008. The next containment concrete inspection is scheduled for 2015 and the next tendon inspection is scheduled for 2012. Corrective actions identified by these inspections are provided with the information.

4.3.1. IWE Examinations

Refueling Outage 1R18 (2006) Containment In-Service Inspection-Metal Liner

Examinations performed for the Containment Liner Plate per Technical Specification Surveillance Procedure RT-142, "Containment Inservice Inspection-Metal Liner," identified several small areas that were recorded as indications on the NDE examination reports during refueling outage 1R18 and documented in the PLP corrective action program (CAP) as AR01022856. In a letter provided by the NDE inspectors, these visual observations were categorized as surface corrosion and further clarified that it was not excessive. The corrosion clarification was validated by performance of ultrasonic test (UT) examinations in several areas that were determined to be representative of all corroded areas in containment. The minimum thickness reported for these areas was 0.234 (nominal 0.250) inches and 0.485 (nominal 0.500) inches. The RT-142 examinations also identified a small area of missing moisture barrier, which was documented in the PLP CAP as AR01024143. This area of the moisture barrier was subsequently replaced as required by the applicable code and successfully re-inspected.

All reported visual observations were considered cosmetic with no areas of suspect damage or deterioration, which would impact the structural integrity or leak tightness of the containment liner. The RT-142 examination of the containment liner plate was successfully completed for refueling outage 1R18 and was "Accepted by Examination" in accordance with applicable requirements.

Refueling Outage 1R20 (2009) Containment In-Service Inspection-Metal Liner

Examinations performed for RT-142 identified only one recordable condition that was not previously evaluated. The recordable condition, documented in the PLP CAP as CR-PLP-2009-01791, was excessive boric acid on penetration number 68. After the boric acid was removed a subsequent inspection was performed and it found penetration number 68 to be acceptable. All the visual observations noted by this inspection were cosmetic with no areas of suspect damage or deterioration, which would impact the structural integrity or leak tightness of the containment liner. The moisture barrier inspection report noted that two areas of the barrier were delaminated, which did not require repairs.

In addition to the containment metal liner inspection performed during refueling outage 1R20, a general visual inspection of all accessible exterior surfaces of the containment was performed per Technical Specification Surveillance Procedure RT-203, "Containment Visual Inspection." The completion of the general visual inspection of the exterior surfaces of the containment resulted in finding no structural problems, which could affect the containment structure leakage integrity. Minor observations were identified and recorded in the inspection report to be monitored and trended in future engineering program inspections.

4.3.2 IWL Examinations

A. IWL (Concrete)

Concrete visual examinations were performed during the summer of 2000 under Technical Specification Surveillance Procedure FT-7, "Containment Visual Inspection," to satisfy TS 4.5 and TS 6.6 requirements. During examination of auxiliary building room no. 232, purification filter room, oil was observed at the tendon buttress. This oil was evidence of grease leakage from one or more tendons at the buttress. During examination of the containment from the tendon access tunnel, grease leakage was discovered at tendons V-176 and V-306; concrete "pop-outs" exposing near surface rebar were discovered near tendons V-128, V-126, V-82 and V-86; a concrete "pop-out," which did not expose rebar, was discovered near tendon V-208. Tendons were not installed at locations V-142 and V-208. These observations were documented in the PLP CAP as CPAL0001422. The missing tendons were documented in the PLP FSAR Section 5.8.2.3. The concrete "pop-outs" discovered in the tendon access tunnel were considered to be surface in nature and did not affect the containment basemat. Historical information documented in the PLP CAP has indicated that grease leakage has not resulted in tendon wire corrosion.

Concrete visual examinations were performed during the month of June, 2005 under FT-7 to satisfy TS Administrative requirements 5.5.5 and 5.6.7. During the examination various minor recordable indications were observed. Examples of these indications included 1) tendon grease at various tendon buttresses, 2) actual grease leakage was observed at tendon caps in the tendon tunnel and on the containment dome, and 3) concrete "pop-outs," "spalls," "cracks," and indications described in visual examination procedure. These observations were entered into the PLP CAP as AR00861254. This action request provided the evaluation of these observations. All of the observations of degradation identified during the performance of Technical Specification Surveillance Procedure FT-7 were found to be minor with no operability concerns. All indications were considered to affect only the outer portions of the concrete structure and were considered to be cosmetic in nature.

Concrete visual examinations were performed during the summer of 2010 under Technical Specification Surveillance Procedure FT-7, "Containment Visual Inspection." Concrete "pop-out," "spalls," and "cracks" were observed and recorded. All newly discovered items documented In accordance with inspection code requirements, indications were reviewed by the Responsible Engineer and determined to support continued containment operability. No conditions in accessible areas indicated the presence of or could reasonably result in degradation of inaccessible areas. It was determined that the containment was fully capable of performing its protective and fission boundary functions.

B. IWL (Tendons)

30-year Tendon Surveillance:

The 30-year tendon surveillance activities were completed on January 13, 2003. The following were the examination and inspection results and corrective actions from this surveillance:

The following tendons exceeded the acceptance criteria for grease replacement:

Dome Tendon D1-38, 18.3 gallons or 32 percent
Vertical Tendon V-16, 11.3 gallons or 14.4 percent
Vertical Tendon V-30, 9.9 gallons or 12.6 percent
Vertical Tendon V-116, 8.8 gallons or 11.2 percent
Vertical Tendon V-330, 8.2 gallons or 10.5 percent

Each of these tendons met the criteria for force measurement, anchorage hardware and surrounding concrete. Tendon wire surfaces were fully covered with a protective grease coating. All locations were refilled by injecting new grease. These tendons were determined to be fully operable.

Grease leakage was discovered at the main gaskets for vertical tendons V-98, V-132, V-134, V-150, V-154, V-178, V-218, V-166, and V-186. All leaks were from the top or shop end on the containment dome. It was determined that heating of the grease, following the filling of the grease cans during the steam generator replacement project, which occurred during a cold weather period in 1991, expanded the grease and pushed the grease by the main gasket. As part of the tendon surveillance project, the main gaskets were replaced and grease leakage from the subject cans stopped. The quantity of grease replacement was sufficient to cover tendon end anchorage hardware but an air pocket was left in the upper portion of the can to allow grease expansion and contraction. The protective grease layer on the tendons was not compromised by the small amount of leakage.

Surveillance activities discovered three missing button heads on vertical tendon V-30, field end. One of the missing button ends was previously recorded during plant construction. The cause of the two additional failed button heads appeared to be fabrication flaws inserted during the button heading process as evidenced by the lack of button heads in the removed grease. There was no visible sign of deterioration at the end of the individual tendon wires. vertical tendon V-30 met all the other applicable tests and inspection criteria.

Surveillance activities discovered a single protruding wire at dome tendon D3-22, shop end. It was determined that the protruding wire in dome tendon D3-22 was due to a break somewhere along the length of the wire as evidenced by all button heads being in place at the field end. Efforts to remove the wire were unsuccessful making it impossible to determine the cause of failure. There was no visible sign of deterioration at the ends of the wires. Tendon force measurement testing was not performed on dome tendon D3-22 due to obstructions. However, all other test and inspection acceptance criteria were met for grease coating, sampling and loss, inspection for water, anchorage corrosion, and concrete inspection.

Water infiltration has been documented during previous tendon surveillance at PLP. The cause of water infiltration has been traced to degraded grease can gaskets and migration through concrete cold joints and tendon sheathing. Surveillance activities discovered 20 ounces of free water at the shop end of dome tendon D1-38. Additionally, the grease sample testing for this tendon indicated chemically combined water at 11 percent at the shop end, only. To fully determine tendon condition, tendon force measure tests and visual exams were performed. Force measurements tests were satisfactory and inspections did not discover any degradation of anchorage components. On the basis of this information, it was concluded that the presence of free water or chemically combined water in the grease was insufficient to cause corrosion or cracking of the anchorage components. Dome tendon D1-38 was refilled by injecting new grease.

In summary, it was concluded, following the 30-year surveillance that the containment structural integrity surveillance program had demonstrated continued containment operability and that the containment structure had not experienced abnormal degradation related to the post-tensioning system.

35-year Tendon Surveillance:

The 35-year tendon surveillance activities were completed on September 6, 2008. The following were the examination and inspection results and corrective actions from this surveillance:

The sheathing filler (grease) samples were tested and found to have acceptable levels of water-soluble ions (chlorides, nitrates and sulfides). All neutralization numbers were above the IWL requirement of 0.0 mg KOH/g and acceptable. Two tendon ends were found with water content above 10% water by weight. The top end of vertical tendon V-212 was found with 14% water by weight and the southeast end of dome tendon D1-38 was found with 28% water by weight. A secondary sample from both of these tendon ends was tested and confirmation was attained. Both tendon ends had acceptable grease coverage and corrosion inspection results. Water content values were below 10% by weight and acceptable for all other samples tested.

Two tendon ends exhibited water during removal of the grease cap; dome tendon D1-38 -23 ounces and dome tendon D1-36 - 1 ounce. No water was exhibited during the inspection of any other tendons. A sample was obtained from dome tendon D1-38 and sent for pH testing. The sample returned with an acceptable pH level of 12.80. A sample was not able to be obtained from dome tendon D1-36 due to the small amount of water present.

Acceptable corrosion levels were found on all tendon ends and no cracks were found on any anchorage components. Cracks in the concrete surrounding the bearing plates were all within allowable tolerance of < 0.010 inch.

All surveillance tendons monitored for forces this inspection period were found to have forces greater than 95% of the corresponding predicted force.

The de-tensioned tendons were re-tensioned with acceptable elongations and acceptable force levels. All test wires removed from de-tensioned tendons were found to have acceptable corrosion levels. All tendon test wire samples had acceptable diameter, yield stress, ultimate stress and elongation results.

All tendons were resealed and re-greased to acceptable levels.

A comparison of as-found force levels to the original force levels was made in an effort to detect any evidence of system degradation. The amount of force loss since the original installation is comparable to the losses of other plants of this age and does not show any evidence of system degradation. Based on the data gathered during the 2008 35-year containment IWL inspection, the conclusion was reached that no abnormal degradation of the post tensioning system had occurred on the PLP containment structure.

4.4 Deficiencies Identified

Consistent with the guidance provided in NEI 94-01, Revision 2, Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE/IWL program examinations or at other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions.

4.5 Plant-Specific Confirmatory Analysis

4.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the PLP ILRT interval from the current ten years to 15 years. This plant-specific risk assessment followed the guidance in NEI 94-01, Revision 2-A, the methodology outlined in EPRI TR-104285, August 1994 and TR-1009325, Revision 2-A, and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implication of corrosion-induced leakage of steel containment liners going undetected during the extended ILRT interval was also used for sensitivity analysis.

The risk assessment performed for the ILRT extension request is based on the current Level 1 PRA model analysis of record. Information developed for the license renewal effort to support the Level 2 release categories is also used in this analysis. Model updates have occurred and are discussed below.

In the June 25, 2008, NRC SE, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. The following Table 4.5-1 SE Section 4.2 limitations and conditions response addresses the ENO response for each of the four limitations and conditions for the use of EPRI TR-1009325, Revision 2 as approved.

Table 4.5-1
SE Section 4.2 Limitations and Conditions Response

Limitations and Conditions (From Section 4.2 of SE)	ENO Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension.	PLP PRA quality is addressed in Section 4.5.2.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or one percent of the total population dose, whichever is restrictive. In addition, a small increase in conditional containment failure probability [CCFP] should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and CCFP acceptance guidelines, and these guidelines have been used for the PLP plant specific assessment.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the PLP plant specific risk assessment.
4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.	PLP analyses do not rely on containment overpressure to assure adequate net positive suction head for ECCS pump following design basis accidents.

4.5.2 PRA Quality

As mentioned earlier, the risk assessment performed for the ILRT extension request is based on the current Level 1 PRA model analysis of record. Information developed for the license renewal effort to support the Level 2 release categories is also used in this analysis. Model updates have occurred and are discussed in Attachment 4.

None of these updates have significantly altered the CDF (core damage frequency) or the LERF (large early release frequency) values such that the bounding analyses performed herein are in question. For this application, the accepted methodology involves a bounding approach to estimate the change in the LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

To address the RG 1.200 requirements; however, Attachment 4, Section A.2, provides a summary of the peer review results from past assessments of the Palisades PRA model. This also includes a report on the latest set of findings from the October 2009 peer review. An evaluation of the impact of these findings on the ILRT extension risk assessment is presented. Attachment 4, Section A.3, provides an assessment of key assumptions and approximations used in this assessment and Attachment 4, Section A.4, summarizes the results of the PRA technical adequacy assessment with respect to this application.

4.5.3 Summary of Plant-Specific Risk Assessment Results

The findings of the PLP risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval from three in ten years to one in 15 years is small. The PLP plant-specific results for extending the ILRT interval to 15 years, taken from Attachment 4, Section 7.0, Conclusions, are summarized below.

1. 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 of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. "Small" changes in risk are defined as increases in CDF below 10^{-5}/yr and increases in LERF below 10^{-6}/yr. Since the ILRT extension was demonstrated to have a negligible impact on CDF for PLP, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included is 2.44E-07/yr, which falls within the small change region of the acceptance guidelines in RG 1.174. In sensitivity analysis using the EPRI Expert Elicitation methodology, the change is estimated as 2.58E-08/yr, which falls within the very small change region.

2. The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all accident sequences, is 1.34E+00 person-rem/yr using the EPRI guidance with the base case corrosion case. The change in dose risk drops to 2.55E-01 person-rem/yr when using the EPRI Expert Elicitation methodology. The value calculated per the EPRI guidance is slightly higher than the acceptance guideline for a "very small" change of ≤1.0 person-rem/yr. However, this calculated increase is conservatively high based on the assignment of the L-LL release category to the intact containment case, which subsequently yields conservative estimates of the EPRI Class 3a and 3b calculated dose results. As such, the risk impact when compared to other severe accident risks is small.
4. The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance is 0.91%, and drops to about 0.10% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.
5. To determine the potential impact from external events, an additional bounding assessment from the risk associated with external events utilizing information from the PLP IPEEE was performed. The total increase in LERF due to internal events and the bounding external events assessment is 4.9E-07/yr, which falls within the small change region of the acceptance guidelines in RG 1.174 and is in Region II of the RG 1.174 acceptance guidelines.
6. Per the Attachment 4, Section 7, Conclusions, the same bounding analysis indicates that the total LERF from both internal and external risks is 1.3E-06/yr, which is less than the RG 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II. This validates that the calculated ΔLERF is acceptable.

Therefore, increasing the ILRT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be significant since it represents only a small change in the PLP risk profile.

Details of the PLP risk assessment are contained in Attachment 4.

4.6 Conclusion

NEI 94-01, Revision 2-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 2-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. ENO is proposing to adopt the guidance of NEI 94-01, Revision 2-A for the PLP 10 CFR 50, Appendix J, testing program plan.

Based on the previous ILRT tests conducted at PLP, it may be concluded that extension of the containment ILRT interval from ten to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B and inspection activities performed as part of the PLP IWE/IWL ISI program.

This experience is supplemented by risk analysis studies, including the PLP risk analysis provided in Attachment 4. The findings of the PLP risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 15 years results in a small change to the PLP risk profile.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

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

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

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

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

NEI 94-01, Revision 2-A, describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. The document incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate test frequencies. In the SE issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions, noted in Section 4.0 of the SE.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized Integrated Leak Rate Test (ILRT) intervals up to 15 years, using current industry performance data and risk-informed guidance. NEI 94-01, Revision 2, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in TR-1009325, Revision 2, for a proposed extension of the ILRT interval to 15 years. In the safety evaluation (SE) issued by NRC letter June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE.

Based on the considerations 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 continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

5.2 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) is proposing a license amendment to the Palisades Nuclear Plant (PLP), Technical Specifications (TS) Section 5.5.14, "Containment Leakage Rate Testing Program." The proposed amendment would replace the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, dated October 2008, as the implementation document used by PLP to develop the PLP performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. The proposed amendment would also extend the interval for the primary containment integrated leak rate test (ILRT), which is required to be performed by 10 CFR 50, Appendix J, from ten years to no longer than 15 years from the last ILRT.

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

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

Response: No.

The proposed amendment involves changes to the PLP containment leakage rate testing program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for development of the PLP performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components would limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 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. ENO has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the PLP performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

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

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

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the PLP performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 2-A.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment would not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current PLP PSA model concluded that extending the ILRT test interval from ten years to 15 years results in a very small change to the PLP risk profile.

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

Based on the above, ENO 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.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

This request is similar in nature to the license amendment authorized by the NRC on March 30, 2010, for the Nine Mile Point Nuclear Station, Unit 2 (TAC No. ME1650, ADAMS Accession Number ML100730032).

TABLE 4.2-2
PALISADES PRESSURE BOUNDARY COMPONENTS SUBJECT TO TYPE B AND TYPE C TESTING

EQUIPMENT ID	DESCRIPTION	PENETRATION TYPE	TEST FREQUENCY (MONTHS)	LAST TEST DATE	NEXT TEST DATE	"AS-LEFT" LEAKAGE (CC/MIN)
MZ-1A	Purge Air Exhaust Penetration	C	60	4/11/09	4/11/14	428.7
MZ-1B	Purge Air Exhaust Bypass Penetration	C	60	4/12/09	4/12/14	405.7
MZ-1C	Purge Air Exhaust Penetration	C	60	4/13/09	4/13/14	0.0
MZ-10	Service Air Piping	C	60	10/24/10	10/24/15	361
MZ-11	Condensate to Shield Cooling Surge Tank	C	60	10/16/10	10/9/15	371
MZ-17	Containment Pressure Instrument	C	60	10/8/10	10/8/15	18
MZ-18	Fuel Transfer Tube Flange	B	30	10/22/10	4/22/13	0.4
MZ-18A	Transfer Tube Winch Cable Flange	B	120	11/6/04	11/06/14	48.8
MZ-19	Personnel Air Lock	B	30	10/16/10	4/16/13	88
MZ-21	Cont H2 Monitoring Return Left Channel	C	60	10/19/10	10/17/15	16.5
MZ-21A	Cont H2 Monitoring Supply Left Channel	C	60	10/20/10	10/18/15	11.2
MZ-25	Clean Waste Receiver Tank Vent to Stack	C	60	4/1/09	4/1/14	59.5
MZ-26 [Note 1]	Nitrogen to Containment	C	30	10/16/10	4/18/13	1360
MZ-27	ILRT Fill Line	C	60	10/12/10	10/12/15	185
MZ-33	Safety Injection Tank Drain	C	60	10/12/10	10/12/15	264
MZ-36	Letdown to Purification Ion Exchanger	C	60	10/12/10	10/9/15	25
MZ-37	Primary System Drain Pump Recirculation	C	60	9/22/07	9/22/12	154.1
MZ-40 [Note2]	Primary Coolant System Sample Line	C	48	10/23/10	10/18/14	0
MZ-40A	Cont H2 Monitoring Return Right Channel	C	60	9/19/07	9/19/12	24.4

TABLE 4.2-2
PALISADES PRESSURE BOUNDARY COMPONENTS SUBJECT TO TYPE B AND TYPE C TESTING

EQUIPMENT ID	DESCRIPTION	PENETRATION TYPE	TEST FREQUENCY (MONTHS)	LAST TEST DATE	NEXT TEST DATE	"AS-LEFT" LEAKAGE (CC/MIN)
MZ-40B	Cont H2 Monitoring Supply Right Chan	C	60	9/19/07	9/19/12	39.1
MZ-41	Degasifier Pump Discharge	C	60	10/19/10	10/19/15	595.6
MZ-42	Deminerlized Water to Quench Tank	C	60	10/16/10	10/16/15	1460
MZ-44	Primary Coolant Pump Controlled Bleed Off	C	60	10/12/07	10/12/12	21.3
MZ-46	Containment Vent Header	C	60	4/14/09	4/02/14	0.0
MZ-47	Primary System Drain Tank Pump Suction	C	60	10/18/10	10/18/15	1090
MZ-48	Cont Isolation & Safety Injection Signal	C	60	3/29/09	3/29/14	0.2
MZ-48	Containment Pressure Instrument Lines Containing Pressure Switches and Transmitters	C	60	3/30/09	3/30/14	75.2
MZ-49	Clean Waste Receiver Tank Pp Suction	C	60	10/23/10	10/21/15	820.6
MZ-50	Escape Air Lock	B	30	8/24/10	02/24/13	27
MZ-51	Equipment Door (Containment Hatch)	B	30	10/26/10	04/26/13	23
MZ-52	Containment Sump Drain to Sump	C	60	4/15/09	4/15/14	48.1
MZ-52A & MZ-17a Note 3	Containment Sump Level Instrumentation	C	30	10/22/10	4/22/13	93
MZ-52B & MZ-56 Note 3	Containment Sump Level Instrumentation	C	30	10/22/10	4/22/13	64
MZ-64	Reactor Cavity Fill and Recirculation	C	60	3/26/09	3/26/14	1190.3
MZ-65 Note 4	Instrument Air	C	60	10/23/10	10/23/15	141
MZ-66 Note 5	ILRT Instrument Line	C	48	10/19/10	10/19/14	1491
MZ-67 Note 6	Clean Waste Receiver Tank Pump Recirculation	C	30	10/23/10	4/23/13	2231

TABLE 4.2-2
PALISADES PRESSURE BOUNDARY COMPONENTS SUBJECT TO TYPE B AND TYPE C TESTING

EQUIPMENT ID	DESCRIPTION	PENETRATION TYPE	TEST FREQUENCY (MONTHS)	LAST TEST DATE	NEXT TEST DATE	"AS-LEFT" LEAKAGE (CC/MIN)
MZ-68	Air Supply to Air Room Penetration	C	60	4/13/09	4/13/14	461.8
MZ-69 Note 7	Clean Waste Receiver Tank Pump Suction	C	30	10/19/10	4/19/13	397.9
MZ-72	Reactor Cavity Drain and Recirculation	C	60	3/26/09	3/25/14	638.3
North Electrical Note 8	North Electrical Penetrations	C	30	10/9/10	4/9/13	3635
South Electrical	South Electrical Penetrations	C	60	10/9/10	10/9/15	163.1

Notes:

1. The as-found leakage recorded for MZ-26, nitrogen to the quench tank, during the 2010 refueling outage exceeded the administrative limit of 2000 cc/min. The performance rating is classified as poor and requires testing on a 30-month frequency.
2. Primary system sample isolation control valve CV-1911 was replaced in the 1995, 1996, 1998, 2001 and 2010 refueling outages. The valve remains on a 48-month test interval even though it meets the requirements for a 60-month test interval because of its previously elevated leak rates.
3. The containment sump water level transmitters of penetrations MZ-52A and MZ-52B were replaced during the 2009 refueling outage which is considered as the replacement of the penetration. The penetration was placed on the base testing interval of 30 months.
4. The IST check valve program requires this test to be performed every other refueling outage to meet closure testing requirements. The 60-month limit ensures that the Appendix J frequency limit is not exceeded.
5. The ILRT instrument line isolation manual valve, MV-VA601 remains on a 48-month test interval even though it meets the requirements for a 60-month test interval because of its previously elevated leak rates.
6. The as-found leakage recorded for MZ-67, clean waste receiver tank pump discharge, during the 2010 refueling outage exceeded the administrative limit of 2000 cc/min. The performance rating is classified as poor and requires testing on a 30-month frequency.
7. MZ-69, clean waste receiver tank pump suction, exceeded the administrative limit in the 2004 refueling outage with resin residue found on a valve seat. The penetration remains on a 30-month test interval even though it meets the requirements for a 60-month test interval because of its previously elevated leak rates.
8. The north electrical penetration has shown an increase in leakage between the 2009 and the 2010 tests. Entergy procedural guidance requires a performance based component is to be tested, at the base interval, if engineering judgment determines a component performance history is invalidated. The 2010 refueling outage test results were within the administrative limit of 5000 sccm and, therefore, do not invalidate the component's performance history. However, it does show an increase in leakage that indicates the penetration should be tested again at a frequency of less than 60 months.

Attachment 2

Renewed Operating License Page Change Instructions

and

Revised Technical Specifications Page

5.0-18

Two pages follow

ATTACHMENT TO LICENSE AMENDMENT NO.

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following pages of Appendix A Technical Specifications and replace with the attached revised pages. The revised pages are identified by amendment number and contain lines in the margin indicating the areas of change.

REMOVE	INSERT
Page 5.0-18	Page 5.0-18

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak 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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Attachment 3

Mark-up of Technical Specifications Page

5.0-18

One page follows

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

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

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1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Attachment 4

Risk Impact Assessment of Extending Palisades ILRT Interval

106 pages follow

Risk Impact Assessment of Extending Palisades ILRT Interval

TABLE OF CONTENTS

List of Tables

1.0 PURPOSE OF ANALYSIS

1.1 Purpose

The purpose of this analysis is to provide an assessment of the risk associated with implementing a permanent extension of the Palisades Nuclear Plant (PLP) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology outlined in EPRI TR-104285 [2], the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [22], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [19]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the October 2008 EPRI final report [22].

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than the normal containment leakage of 1.0La (allowable leakage).

The basis for a 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 [5], "Performance-Based Containment Leak Test Program," provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [2].

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for PLP.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 [2] methodology to perform the risk assessment. In October 2008, EPRI TR-1018243 [22] was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174 [4]. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas TR-104285 considered only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for PLP employs the EPRI TR-1018243 methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary.

1.3 Acceptance Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Note that a separate discussion in Section 5.8 confirms that the CDF change is bounded by the calculated LERF change for PLP. Therefore, since the Type A test does not significantly impact CDF for PLP, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year, provided that the total from all contributors (including external events) can be reasonably shown to be less than 10^{-5} per reactor year. RG 1.174 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 (CCFP) is also calculated to help ensure that the defense-in-depth philosophy is maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of [22]) indicate a range of incremental increases in population dose1 that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase of ≤ 1.0 person-rem per year, or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals.

1 The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, whereas this analysis uses 100La.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI methodology is used for evaluating the change in risk associated with increasing the test interval to fifteen years [22]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current PLP PRA analysis of record and the subsequent containment responses for the various fission product release categories including the release size.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report [22].
2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 and compare this change with the acceptance guidelines of RG 1.174 [4].
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

Furthermore,

- Consistent with the other industry containment leak risk assessments, the PLP assessment uses population dose as one of the risk measures. The other risk measures used in the PLP assessment are the conditional containment failure probability (CCFP) for defense-in-depth considerations, and LERF to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for PLP uses ground rules and methods to calculate changes in the above risk metrics that are consistent with those outlined in the current EPRI methodology [22].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The PLP Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the PLP internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if external events were to be included in the calculations; however, external events have been accounted for in the analysis based on the available information from the PLP IPEEE [18] as described in Section 5.7.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in the PLP Environmental Report for License Renewal (Attachment E, Severe Accident Mitigation Alternatives) [9].
- Accident classes describing radionuclide release end states and their definitions are consistent with the EPRI methodology [22] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is 100La, based on the recommendations in the latest EPRI report [22]. It should be noted that most previous industry ILRT extension requests utilized 35La.
- Based on the EPRI methodology, the Class 3b sequences can be conservatively categorized as LERF and the increase used as a surrogate for LERF. However, in this analysis, the releases associated with a 100La release would not necessarily be consistent with a "Large" release for PLP [22].
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of the estimated 2031 population data from the License Renewal Application [9] is appropriate for this analysis.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].

4.0 INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources Available

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [8]
6. NUREG-1493 [5]
7. EPRI TR-104285 [2]
8. Calvert Cliffs liner corrosion analysis [19]
9. EPRI 1018243 [22]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the last study complements the previous EPRI report [2] and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

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

NUREG-1273 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [8]

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

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1E-8/yr to 1E-7/yr) is realized from extending the test intervals from 3 per 10 years to 1 per 10 years.

NUREG-1493 [5]

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

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 [14] Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 [5] 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 sequences into eight categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment 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..."

Release Category Definitions

Table 4.1-1 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [22]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4.1-1
EPRI/NEI Containment Failure Classifications [22]

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 L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in 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 their 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) program.
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.

Calvert Cliffs Liner Corrosion Analysis [19]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. PLP has a similar type of containment.

EPRI 1018243 [22]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years. This risk impact assessment complements the previous EPRI report, TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals. The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment considers the change in risk based on population dose, whereas the revision considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry and regulatory guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffrey's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the PLP analysis presented here in Section 6.2.

4.2 Plant-Specific Inputs

The PLP specific information used to perform this ILRT interval extension risk assessment includes the following:

- PRA model quantification results [17]
- Population within a 50-mile radius [9]

PLP Internal Events PRA Model

The current PLP Internal Events PRA analysis of record is an event tree / linked fault tree model characteristic of the as-built, as-operated plant. Based on the uncertainty analysis results found in Table 7.2-2 of Reference [17], the mean value of the distribution for internal events core damage frequency (CDF) is 2.66E-05/yr.

PLP Internal Events Release Categories

The Level 2 release category frequencies were developed from the relative contributions to CDF for those analyzed containment failure modes that were documented in Table 7.1 of Reference [25]. Table 4.2-1 summarizes the pertinent PLP results in terms of end-states where a representative release category is assigned for each end-state. The total Large Early Release Frequency (LERF) in Table 4.2-1 was found to be 4.19E-07/yr, with the total of all release category frequencies, including the intact case, being set equal to the mean CDF value of 2.66E-05/yr. The individual release category frequencies are utilized here to provide the necessary delineation for the ILRT risk assessment with the corresponding EPRI class for each release category being listed in Table 4.2-1. A discussion of the release categories follows this table.

Table 4.2-1
Population Dose and Dose Risk Within 50 Miles for PLP Organized by Containment Failure Mode

			Column Identifier =>	1	2	3	4	
Containment Failure Mode # Ref. [25]	**Containment Failure Mode Description Ref. [25]**	**Rep. Release Category Ref. [9]**	**EPRI Release Category Ref. [22]**	**Fractional Contribution to CDF Ref. [25]**	**Calculated Frequency (1/yr) (Col. 1 x CDF)**	**Population Dose Based on Release Category (p-rem) (Based on Dose in sieverts from Table E.3-4 of Ref. [9])**	**Dose Risk (p-rem/yr) (Col. 2 x Col. 3)**	**LERF (Y or N)**
2	Containment Bypass (Transient induced SGTR events)	E-M	8	8.85E-04	2.35E-08	5.68E+05	1.34E-02	N
9	Containment Bypass (SGTR Initiating Events)	I-H	8	1.60E-01	4.26E-06	1.85E+06	7.87E+00	N
1	Containment Bypass (ISLOCA)	E-H	8	3.88E-09	1.03E-13	6.15E+06	6.35E-07	Y
4	Containment Failed Early (DCH, flammable gas burns, steam explosions)	E-H	7	1.08E-02	2.88E-07	6.15E+06	1.77E+00	Y
7	Intermediate Containment Failure > 11 hours (flammable gas burns, steam and non-condensable gas generation due to core-concrete interaction)	I-M	7	0.00E+00	0.00E+00	1.33E+06	0.00E+00	N

Table 4.2-1
Population Dose and Dose Risk Within 50 Miles for PLP Organized by Containment Failure Mode

			Column Identifier =>	1	2	3	4	
Containment Failure Mode # Ref. [25]	**Containment Failure Mode Description Ref. [25]**	**Rep. Release Category Ref. [9]**	**EPRI Release Category Ref. [22]**	**Fractional Contribution to CDF Ref. [25]**	**Calculated Frequency (1/yr) (Col. 1 x CDF)**	**Population Dose Based on Release Category (p-rem) (Based on Dose in sieverts from Table E.3-4 of Ref. [9])**	**Dose Risk (p-rem/yr) (Col. 2 x Col. 3)**	**LERF (Y or N)**
5	Containment Failed Late > 40 hours (flammable gas burns, steam and non-condensable gas generation due to core-concrete interaction)	L-L	7	9.95E-02	2.65E-06	6.54E+04	1.73E-01	N
6	Containment Intact	L-LL	1	3.39E-01	9.01E-06	4.10E+04	3.69E-01	N
8	Intermediate Containment Failure > 4 hours (Core to the Aux. Building > 4 hours)	I-M	7	3.85E-01	1.02E-05	1.33E+06	1.36E+01	N
3	Containment Isolation Failure	E-H	2	4.93E-03	1.31E-07	6.15E+06	8.05E-01	Y
			Totals:	1.00	2.66E-05		24.645	

Detailed Release Categories

The release categories considered the magnitude of the radionuclide release, e.g., concentration of cesium iodide (CsI), and the time of the release. Table 4.2-2 shows how the different release categories were organized based on a two-term matrix (i.e., severity and time).

Table 4.2-2
Release Category Definitions [9]

Release Timing		**Release Severity Source Term Release Fraction**	
Classification Category	Time of Release (noble gases or CsI)	Classification Category	Percent CsI in Release
Late (L)	> 24 hours	High (H)	> 10
Intermediate (I)	4 to 24 hours	Moderate (M)	1 to 10
Early (E)	< 4 hours	Low (L)	0.1 to 1
		Low-Low (LL)	< 0.1

Release Category Early-High (E-H)

This release category represents those early large releases that are characteristic contributors to LERF, such as containment bypass scenarios that would include an interfacing system LOCA (ISLOCA). Radionuclides are assumed to be released from the primary system directly to the environment. Onset of release is taken when the core is assumed to uncover at 1.3 hours, since the containment is already bypassed at that time. All of the safety injection is assumed to fail when the safety injection and refueling water tank (SIRWT) is depleted at 0.44 hrs.

Release Category Early-Moderate (E-M)

This release category captures those sequences that yield an early release with moderate radionuclide content. A transient accident with an induced steam generator tube rupture (SGTR) is a representative case for this type of release category.

Release Category Intermediate-High (I-H)

This release category captures those sequences that yield an intermediate release with high radionuclide content. A SGTR with a stuck open relief valve on the faulted steam generator is a representative case for this type of release category.

Release Category Intermediate-Moderate (I-M)

This release category represents those sequences with reactor vessel failure at low pressure, no upward debris dispersal and debris relocation to the Auxiliary Building. Prior to modification of the containment sump, this would have represented an early release, but given that the sump modification at Palisades results in a delayed failure of containment, the timing is shifted to intermediate. The release frequency calculated for this category of accidents in Reference [25] proved to be zero.

Release Category Late-Low (L-L)

This release category represents those sequences with reactor vessel failure at high pressure, upward debris dispersal, late containment failure and no core-concrete interaction due to debris cooling.

Release Category Late-Low Low (L-LL)

This release category is similar to the L-L category described above except that a late revaporization release from the primary system does not occur. The intact containment case was also conservatively binned to this release category. This conservatism is noteworthy as it will also impact the doses assigned to the EPRI Class 3 release categories.

Population Dose Risk Calculations

The population dose that was calculated in the PLP SAMA Evaluation [9] for each of the release categories was used to obtain a population dose risk for this ILRT analysis using the Level 2 release category frequencies calculated in Reference [25]. The population dose risk (Column 4 of Table 4.2-1) was found by multiplying the release category frequency (Column 2 of Table 4.2-1) by the associated population dose (Column 3 of Table 4.2-1).

Table 4.2-3 lists the population dose risk and average population dose organized by EPRI release category, including the delineation of LERF and non-LERF frequencies for classes 7 and 8.

Table 4.2-3
PLP Population Dose Risk and Averaged Population Dose Organized by EPRI Release Category

EPRI RELEASE CATEGORY	POPULATION DOSE RISK (PERSON-REM/YR)	RELEASE FREQUENCY (1/YR)	AVERAGED POPULATION DOSE (PERSON-REM)(1)
1	3.69E-01	9.01E-06	4.10E+04
2	8.06E-01	1.31E-07	6.15E+06
7 non-LERF	1.38E+01	1.29E-05	1.07E+06
7 LERF	1.77E+00	2.88E-07	6.15E+06
8 non-LERF	7.89E+00	4.28E-06	1.84E+06
8 LERF	6.36E-07	1.03E-13	6.15E+06
Total	2.46E+01	2.66E-05	

(1) Obtained by dividing the population dose risk shown in the second column by the release category frequency in the third column of this table.

The frequencies for the severe accident classes defined in Table 4.1-1 are developed for PLP based on the assignments shown above in Table 4.2-3, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments are made to the Class 3b as well as Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the latest EPRI guidance [22], as the mean failure estimated from the available data (i.e., 2 "small" failures that could only have been discovered by the ILRT in 217 tests leads to a 2/217=0.0092 mean value). For Class 3b, consistent with latest available EPRI data

[22], a non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

The EPRI methodology contains information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174 [22]. This information includes a discussion of conservatisms in the quantitative guidance for delta LERF. EPRI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The methodology states:

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

The application of this additional guidance to the analysis for PLP (as detailed in Section 5) means that the Class 2 and Class 8 LERF sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. In general, Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events that contribute to LERF.

Consistent with the EPRI methodology [22], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing, given a 10-year vs. a 3-yr interval. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

PLP Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart) where the calculated performance leakage rate was less than 1.0La, and in compliance with the performance factors in NEI 94-01, Section 11.3. Based on the successful completion of two consecutive ILRTs at PLP, the current ILRT interval is once per ten years (with a current extension of 15 months in effect). Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry-wide historical results as noted in the EPRI guidance document [22].

EPRI Methodology

This analysis uses the approach outlined in the EPRI Methodology [22]. The six steps of the methodology are:

1. Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.
2. Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.
3. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
4. Determine the risk impact in terms of the change in LERF and the change in CCFP.
5. Consider both internal and external events.
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis.

The first three steps of the methodology deal with calculating the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF for PLP, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The fourth step of the methodology also calculates the change in containment failure probability, referred to as the conditional containment failure probability, CCFP. The NRC has previously accepted similar calculations [7] for the change in CCFP as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such, this step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174. Step 5 takes into consideration the additional risk due to external events, and Step 6 investigates the impact on results due to varying the assumptions associated with the liner corrosion rate and failure to visually identify pre-existing flaws.

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

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [19]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. PLP has a similar containment type.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

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

Assumptions

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- The two corrosion events over a 5.5 year data period are used to estimate the liner flaw probability in the Calvert Cliffs analysis and are assumed to be applicable to the Palisades containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner. It is noted that two additional events have occurred in recent years (based on a data search covering approximately 9 years documented in Reference [24]). In November 2006, the Turkey Point 4 containment building liner developed a hole when a sump pump support plate was moved. In May 2009, a hole approximately 3/8" by 1" in size was identified in the Beaver Valley 1 containment liner. For risk evaluation purposes, these two more recent events occurring over a 9 year period are judged to be adequately represented by the two events in the 5.5 year period of the Calvert Cliffs analysis incorporated in the EPRI guidance (See Table 4.4-1, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages (See Table 4.4-1, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every two years and every ten years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome region, and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For Palisades, the containment failure probabilities are less than these values at 55 psig, which is the containment design pressure as reported in the IPE submittal [16]. The probabilities of 1% for the cylinder and dome, and 0.1% for the basemat, albeit conservative, are used in this analysis. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4.4-1, Step 4).
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used for the containment cylinder and dome. For the containment basemat, 100% is assumed unavailable for visual inspection. To date, all liner corrosion events have been detected through visual inspection (See Table 4.4-1, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 4.4-1
Steel Liner Corrosion Base Case

<table>
<tr><th>STEP</th><th>DESCRIPTION</th><th colspan="2">CONTAINMENT CYLINDER AND DOME</th><th colspan="2">CONTAINMENT BASEMAT</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood
Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).</td><td colspan="2">Events: 2

2/(70 * 5.5) = 5.2E-3</td><td colspan="2">Events: 0 (assume half a failure)
0.5/(70 * 5.5) = 1.3E-3</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Age Adjusted Steel Liner Flaw Likelihood
During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).</td><td><u>Year</u>
1
avg 5-10
15</td><td><u>Failure Rate</u>
2.1E-3
5.2E-3
1.4E-2</td><td><u>Year</u>
1
avg 5-10
15</td><td><u>Failure Rate</u>
5.0E-4
1.3E-3
3.5E-3</td></tr>
<tr><td colspan="2">15 year average = 6.27E-3</td><td colspan="2">15 year average = 1.57E-3</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years
Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis - See Table 6 of Reference [19]).</td><td colspan="2">0.71% (1 to 3 years)
4.06% (1 to 10 years)
9.40% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, the values are calculated based on the 3, 10, and 15 year intervals.)</td><td colspan="2">0.18% (1 to 3 years)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, values are calculated based on the 3, 10, and 15 year intervals.)</td></tr>
</table>

Table 4.4-1
Steel Liner Corrosion Base Case

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1% (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.00071% (at 3 years)** =0.71% * 1% * 10% **0.00406% (at 10 years)** =4.06% * 1% * 10% **0.0094% (at 15 years)** =9.40% * 1% * 10%	**0.00018% (at 3 years)** =0.18% * 0.1% * 100% **0.00102% (at 10 years)** =1.02% * 0.1% * 100% **0.00235% (at 15 years)** =2.35% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome, and the containment basemat:

At 3 years : 0.00071% + 0.00018% = 0.00089%
At 10 years: 0.00406% + 0.00102% = 0.00508%
At 15 years: 0.0094% + 0.00235% = 0.01175%

5.0 RESULTS

The application of the approach based on EPRI Guidance [22] has led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

The analysis performed examined PLP-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the categorization of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI 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, if applicable. (EPRI 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 Class 6 sequences). Consistent with the EPRI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypass (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5.0-1
Accident Classes

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal -Type B)
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 (SGTR and Interfacing System LOCA)
CDF	**All CET End states (including very low and no release)**

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

Step 1 Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5.0-1.

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

Step 3 Evaluate risk impact of extending Type A test interval from 3 to 15 and 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 the impact on the Conditional Containment Failure Probability (CCFP).

5.1 Step 1 – Quantify the Base-Line Risk in Terms of Frequency per Reactor Year

This step involves the review of the PLP Level 2 release category frequency results [17,25]. As described in Section 4.2, the release categories were assigned to the EPRI classes as shown in Table 4.2-1. This application combined with the PLP dose risk (person-rem/yr) also shown in Table 4.2-1 forms the basis for estimating the increase in population dose risk.

For the assessment of the impact on the risk profile due to the ILRT extension, the potential for pre-existing leaks is included in the model. These pre-existing leak events are represented by the Class 3 sequences in EPRI TR-1018243 [22]. Two failure modes were considered for the Class 3 sequences, namely Class 3a (small breach) and Class 3b (large breach).

The set of containment release frequencies from the PLP Level 2 PRA model [17,25] and the population dose from the SAMA evaluation [9] were used to develop the updated population dose and dose risk shown in Table 4.2-3 that is consistent with the EPRI categories listed in Table 5.0-1.

<u>Class 1 Sequences</u>
This group represents the frequency when the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 8.71E-06/yr (refer to Table 5.1-1, containment release type 1) and is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF. For this analysis, the associated maximum containment leakage for this group is 1La, consistent with an intact containment evaluation.

<u>Class 2 Sequences</u>
This group consists of large containment isolation failures. For PLP, this frequency is 1.31E-07/yr (refer to Table 5.1-1, containment release type 2).

<u>Class 3 Sequences</u>
This group represents pre-existing leakage in the containment structure (e.g., containment liner). The containment leakage for these sequences can be either small (2La to 100La) or large (>100La). In this analysis, a value of 10La was used for small pre-existing flaws and 100La for relatively large flaws.

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 4.3)
$PROB_{Class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 4.3)

As described in Section 4.3, additional consideration is made to not apply these failure probabilities to those cases that are already considered LERF scenarios (i.e., the Class 2 and Class 7 and 8 LERF contributions).

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7 LERF + Class 8 LERF)]
= 0.0092 * [2.66E-05 – (1.31E-07 + 2.88E-07 + 1.03E-13)]
= 2.41E-07/yr
Class_3b = 0.0023 * [CDF – (Class 2 + Class 7 LERF + Class 8 LERF)]
= 0.0023 * [2.66E-05 – (1.31E-07 + 2.88E-07 + 1.03E-13)]
= 6.02E-08/yr

For this analysis, the associated containment leakage for Class 3a is 10La and 100La for Class 3b, which is consistent with the latest EPRI methodology [22].

Class 4 Sequences

This group represents containment isolation failure-to-seal of Type B test components. 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 this analysis.

Class 5 Sequences

This group represents containment isolation failure-to-seal of Type C test components. Because these 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 with a failure-to-seal containment leakage due to failure to isolate the containment. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the EPRI guidance, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences

This group represents containment failure induced by severe accident phenomena. For PLP, the frequency for non-LERF Class 7 sequences is 1.29E-05/yr, and for LERF Class 7 sequences, the total is 2.88E-07.

Class 8 Sequences

This group represents sequences where containment bypass occurs. The failure frequency for non-LERF Class 8 sequences is 4.28E-06/yr, and for LERF Class 8 sequences, the total is 1.03E-13/yr.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI TR-1018243 [22] and are shown in Table 5.1-1.

Table 5.1-1
Radionuclide Release Frequencies As A Function Of Accident Class (PLP Base Case)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)
1	No Containment Failure	8.71E-06
2	Large Isolation Failures (Failure to Close)	1.31E-07
3a	Small Isolation Failures (liner breach)	2.41E-07
3b	Large Isolation Failures (liner breach)	6.02E-08
4	Small Isolation Failures (Failure to seal -Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.29E-05
7 LERF	Failures Induced by Phenomena (LERF)	2.88E-07
8 non-LERF	Containment Bypass (non-LERF)	4.28E-06
8 LERF	Containment Bypass (Interfacing System LOCA)	1.03E-13
CDF	**All CET End states (including intact case)**	**2.66E-05**

5.2 Step 2 – Develop Plant-Specific Person-REM Dose (Population Dose) per Reactor Year

Plant-specific release analyses were performed to estimate the weighted average person-rem doses to the population within a 50-mile radius from the plant. The releases are based on a combination of the information provided by the PLP SAMA analysis [9] and the Level 2 containment failure release frequencies [17,25] (see Table 4.2-3 of this analysis). The results of applying these releases to the EPRI containment failure classifications are summarized as follows:

Class 1 = 4.10E+04 person-rem (at 1.0La)
Class 2 = 6.15E+06 person-rem
Class 3a = 4.10E+04 person-rem x 10La = 4.10E+05 person-rem
Class 3b = 4.10E+04 person-rem x 100La = 4.10E+06 person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7 non-LERF = 1.07E+06 person-rem
Class 7 LERF = 6.15E+06 person-rem
Class 8 non-LERF = 1.84E+06 person-rem
Class 8 LERF = 6.15E+06 person-rem

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [22] for all EPRI classes are provided in Table 5.2-1, which includes the values previously presented in Table 4.2-3 as well as the Class 3a and 3b population doses calculated above.

Table 5.2-1
PLP Population Dose
for Population Within 50 Miles

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)
1	No Containment Failure (1 La)	4.10E+04
2	Large Isolation Failures (Failure to Close)	6.15E+06
3a	Small Isolation Failures (liner breach)	4.10E+05
3b	Large Isolation Failures (liner breach)	4.10E+06
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06
7 LERF	Failures Induced by Phenomena (LERF)	6.15E+06
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06
8 LERF	Containment Bypass (LERF)	6.15E+06

The above population dose, when multiplied by the frequency results presented in Table 5.1-1, yields the PLP baseline mean dose risk for each EPRI accident class. These results are presented in Table 5.2-2.

Table 5.2-2
PLP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 3 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure (2)	4.10E+04	8.71E-06	3.57E-01	8.71E-06	3.57E-01	--
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	4.10E+05	2.41E-07	9.88E-02	2.41E-07	9.88E-02	--
3b	Large Isolation Failures (liner breach)	4.10E+06	6.02E-08	2.47E-01	6.04E-08	2.48E-01	1E-3
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.2-2
PLP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 3 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		**2.66E-05**	**2.497E+01**	**2.66E-05**	**2.497E+01**	**1E-3**

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

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

The baseline PLP dose compares favorably with other plants given the relative population densities surrounding each location:

PLANT	ANNUAL DOSE (PERSON-REM/YR)	REFERENCE
Indian Point 3	14,515	[7]
Peach Bottom	6.2	[21]
Farley Unit 2	2.4	[23]
Farley Unit 1	1.5	[23]
Crystal River	1.4	[20]
Palisades	25.0	[Table 5.2-2]

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

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 year interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the EPRI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1 for PLP.

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 not detecting a leak in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-2.

Table 5.3-1
PLP Annual Dose As A Function Of Accident Class; Characteristic Of Conditions For 1 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure (2)	4.10E+04	8.00E-06	3.28E-01	8.00E-06	3.28E-01	--
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	4.10E+05	8.02E-07	3.29E-01	8.02E-07	3.29E-01	--
3b	Large Isolation Failures (liner breach)	4.10E+06	2.01E-07	8.22E-01	2.02E-07	8.28E-01	5.5E-3
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.3-1
PLP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 10 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		**2.66E-05**	**2.575E+01**	**2.66E-05**	**2.575E+01**	**5.5E-3**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

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

Table 5.3-2
PLP Annual Dose As A Function Of Accident Class; Characteristic Of Conditions For 1 in 15 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	No Containment Failure (2)	4.10E+04	7.50E-06	3.076E-01	7.50E-06	3.074E-01	-2E-04
2	Large Isolation Failures (Failure to Close)	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	--
3a	Small Isolation Failures (liner breach)	4.10E+05	1.20E-06	4.94E-01	1.20E-06	4.94E-01	--
3b	Large Isolation Failures (liner breach)	4.10E+06	3.01E-07	1.234E+00	3.04E-07	1.247E+00	1.3E-2
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A

Table 5.3-2
PLP Annual Dose As A Function Of Accident Class;
Characteristic Of Conditions For 1 in 15 Year ILRT Frequency

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7 non-LERF	Failures Induced by Phenomena (non-LERF)	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	--
7 LERF	Failures Induced by Phenomena (non-LERF)	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	--
8 non-LERF	Containment Bypass (non-LERF)	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	--
8 LERF	Containment Bypass (LERF)	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	--
CDF	**All CET end states**		**2.66E-05**	**2.630E+01**	**2.66E-05**	**2.631E+01**	**1E-2**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years.

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

5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr, and small changes in LERF as below 1E-6/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For PLP, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on the original 3-in-10 year test interval assessment from Table 5.2-2, the Class 3b frequency is 6.04E-08/yr, which includes the corrosion effect of the containment liner. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 2.02E-07/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 3.04E-07/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years (including corrosion effects) is 2.4E-07/yr. Similarly, the increase due to increasing the interval from 10 to 15 years (including corrosion effects) is 1.0E-07/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is within Region II of Figure 4 of Reference [4] (small changes in LERF) when comparing the 15 year results to the original 3-in-10 year requirement.

5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability

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

The change in CCFP can be calculated by using the method specified in the EPRI methodology [22]. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The following table shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects in which the flaw rate is assumed to double every five years.

CCFP 3 IN 10 YRS	CCFP 1 IN 10 YRS	CCFP 1 IN 15 YRS	$\Delta CCFP_{15-3}$	$\Delta CCFP_{15-10}$
66.37%	66.90%	67.28%	0.91%	0.38%

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

The change in CCFP of approximately 1% as a result of extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be relatively insignificant.

5.6 Summary of Internal Events Results

Table 5.6-1 summarizes the internal events results of this ILRT extension risk assessment for PLP.

Table 5.6-1
PLP ILRT Cases:
Base, 3 to 10, and 3 to 15 Yr Extensions
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	4.10E+04	8.71E-06	3.57E-01	8.00E-06	3.28E-01	7.50E-06	3.074E-01
2	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	1.31E-07	8.06E-01
3a	4.10E+05	2.41E-07	9.88E-02	8.02E-07	3.29E-01	1.20E-06	4.94E-01
3b	4.10E+06	6.04E-08	2.48E-01	2.02E-07	8.28E-01	3.04E-07	1.247E+00
7 non-LERF	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	1.29E-05	1.38E+01
7 LERF	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	2.88E-07	1.77E+00
8 non-LERF	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	4.28E-06	7.89E+00
8 LERF	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	1.03E-13	6.36E-07
Total		2.66E-05	2.497E+01	2.66E-05	2.575E+01	2.66E-05	2.631E+01
ILRT Dose Rate from 3a and 3b		3.47E-01		1.16E+00		1.74E+00	
Delta Total Dose Rate[1]	From 3 yr	---		7.81E-01		1.34E+00	
	From 10 yr	---		---		5.64E-01	
3b Frequency (LERF)		6.04E-08		2.02E-07		3.04E-07	
Delta 3b LERF	From 3 yr	---		1.42E-07		2.44E-07	
	From 10 yr	---		---		1.02E-07	
CCFP %		66.37%		66.90%		67.28%	
Delta CCFP %	From 3 yr	---		0.53%		0.91%	
	From 10 yr	---		---		0.38%	

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

5.7 External Events Contribution

Since the risk acceptance guidelines in RG 1.174 are intended for comparison with a full-scope assessment of risk, including internal and external events, a bounding analysis of the potential impact from external events is presented here.

The method chosen to account for external events contributions is similar to that used in the SAMA analysis [9] in which a multiplier was applied to the internal events results based on the IPEEE methodology [18]. The contributions of the external events from the original IPEEE analysis are summarized in Table 5.7-1.

Table 5.7-1
Original IPEEE Contributor Summary [18]

EXTERNAL EVENT INITIATOR GROUP	CDF (1/YR)
Seismic	8.88E-06
Internal Fire	3.31E-05
High Winds	N/A (screened per NUREG-1407 and GL 88-20)
External Floods	N/A (screened per NUREG-1407 and GL 88-20)
Transportation and Nearby Facility Accidents	N/A (screened per NUREG-1407 and GL 88-20)
Total (for initiators with CDF available)	**4.20E-05/yr**

The CDF due to fire was calculated based on the total of all the fire scenarios modeled in the IPEEE. The fire analysis included a number of conservative assumptions. For example, automatic or manual fire suppression was not credited except in the control room, cable spreading room and Class 1E switchgear rooms. Even when suppression was credited, the AFW system was assumed failed due to the fire. Fires were also assumed to completely engulf an area, once ignited, and fail all equipment and cabling within the fire area/zone if not suppressed. The fire analysis performed for the IPEEE began in 1994 and reflected some of the major plant changes made since the IPE [16]. The internal fire assessment combined the PRA approach in the IPE with the deterministic evaluation techniques of the Fire Induced Vulnerabilities Evaluation (FIVE) Methodology [26]. This fire analysis has not been updated since the IPEEE submittal.

Since the IPEEE, various improvements and enhancements to the model have been made and the total internal events CDF has been reduced from 5.15E-05/yr [18] to the current uncertainty distribution mean value of 2.66E-05/yr [17].

For the current analysis of record for the Palisades internal events PRA model, Small Break LOCAs and SGTR initiating events are dominant contributors, accounting for over half of the total CDF. Early containment failure sequences, whose frequency would not be increased due to fire events, account for over half of the total LERF [17].

In any event, in addition to modeling limitations, the fire PRA may be subject to more modeling uncertainty than the internal events PRA evaluations. While the fire PRA is generally self-consistent within its calculational framework, the fire PRA CDF results do not compare well with internal events PRAs because of the number of conservative assumptions that have been included in the fire PRA process. Therefore, direct use of the fire PRA results as a reflection of CDF may be inappropriate, and the actual fire CDF based on the IPEEE may be overestimated.

From Table 5.7-1, the external events multiplier can be conservatively calculated by assuming that the external events CDF is equivalent to the internal events CDF, i.e., a multiplier of 1.0. Also, a multiplier of 1.0 is also consistent with the arguments provided in the previously submitted SAMA analysis [9].

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 as 6.04E-08/yr, 2.02E-07/yr, and 3.04E-07/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazards risk, is estimated as shown in Table 5.7-2.

Table 5.7-2
PLP 3b (LERF) as a Function of ILRT Frequency
for Internal and External Events
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	**3B FREQUENCY (1-PER-10 YEAR ILRT)**	**3B FREQUENCY (1-PER-15 YEAR ILRT)**	**LERF INCREASE[(1)]**
Internal Events Contribution	6.04E-08	2.02E-07	3.04E-07	2.44E-07
External Events Contribution (Internal Events CDF x 1.0)	6.04E-08	2.02E-07	3.04E-07	2.44E-07
Combined (Internal + External)	1.21E-07	4.04E-07	6.08E-7	4.88E-07

(1) Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

Thus, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution is estimated as 4.9E-07/yr, which includes the age adjusted steel liner corrosion likelihood.

NRC Regulatory Guide 1.174 [4], "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides NRC recommendations for using risk information in support of applications requesting changes to the licensing basis of the plant. As discussed in Section 2 of this PRA analysis, the risk acceptance criteria of RG 1.174 are used here to assess the ILRT interval extension.

The 4.9E-07/yr increase in LERF due to the combined internal and external events from extending the PLP ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1E-7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-1, the total LERF due to postulated internal event accidents is the sum of the LERF release categories, which is 4.19E-07/yr. Although some of the LERF contributors may not be applicable to external events initiators, the base LERF due to external events is assumed to be the same as the internal events contribution.

Table 5.7-3
Impact of 15-yr ILRT Extension on LERF (3b)

Internal Events LERF	4.19E-07/yr
External Events LERF	4.19E-07/yr
Internal Events LERF due to ILRT (at 15 years) (1)	3.04E-07/yr
External Events LERF due to ILRT (at 15 years) (1)	3.04E-07/yr
Total	**1.4E-06/yr**

(1) Including age adjusted steel liner corrosion likelihood.

As can be seen, the estimated upper bound LERF for PLP is estimated as 1.4E-06/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1E-5/yr.

5.8 CONTAINMENT OVERPRESSURE IMPACTS ON CDF

For Palisades, the design basis calculations for ECCS injection do not rely on containment overpressure. However, the recirculation alignment does rely on HPSI sub-cooling valves and pumps to fully negate the need for containment overpressure. Therefore, in the PRA model the failure of the containment isolation system is included in combination with failure of both HPSI trains aligned with the sub-cooling valves to support HPSI recirculation from the sump. The impact of the ILRT extension on this function can easily be shown to be bounded by the calculated impacts on LERF such that a more detailed assessment does not need to be performed.

1) The scenarios of interest include all LOCA contributors that require recirculation from the sump to ensure success to avoid core damage. This is largely dominated by the small break LOCA contribution of 2.3E-3/yr and therefore can be bounded with a value of approximately 2.5E-3/yr to include all other applicable initiating events and consequential failures.

2) The containment isolation failure probability that leads to loss of containment overpressure can be assumed to be represented by the EPRI Class 3b contribution above. This is conservative since the 100 La leakage rate from the EPRI Class 3b scenarios is likely not to be of sufficient size to actually threaten the development of containment overpressure. The representative Class 3b value is 2.3E-3 and is increased by a factor of five to represent the impacts of the ILRT extension to 15 years.

3) A review of cutsets indicates that the failure of both HPSI sub-cooling trains can be approximated with a total sum value of <1.0E-3 which is dominated by common cause failures of the valves. Therefore, for this assessment a value of 1E-3 can be utilized to show that the CDF impacts are bounded by the calculated LERF impacts.

The information in items 1-3 above can be combined to provide an upper bound estimate of the potential impacts on CDF due to the ILRT extension for PLP as shown below.

[Applicable initiators] * [Containment Isolation Failures] * [HPSI Subcooling Fails]

[2.5E-3/yr] * [2.3E-3] * [1E-3] = 5.8E-9 / yr

The value of 5.8E-9 / yr is much less than the base LERF from EPRI Class 3b of 6.04E-8 /yr shown in Table 5.6-1 above. Since the acceptance criteria for CDF is an order of magnitude higher than the LERF acceptance criteria, then a more detailed CDF assessment does not need to be performed. In any event, if the value calculated above is multiplied by a factor of 5 consistent with the ILRT extension methodology to account for the impacts of the ILRT extension, the calculated bounding CDF from these scenarios is just 2.9E-8/yr. This is well below the acceptance guidelines from RG 1.174 for "very small" changes in CDF and confirms that the impact on CDF from the ILRT extension is negligible.

6.0 SENSITIVITIES

6.1 Sensitivity to Corrosion Impact Assumptions

The results in Tables 5.2-2, 5.3-1, and 5.3-2 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment. In any event, sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder, dome and basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case, the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 9.10E-8/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1-1
Steel Liner Corrosion Sensitivity Cases

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT EXTENSION FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (1.0% Cylinder-Dome, 0.1% Basemat)	Base Case (10% Cylinder-Dome, 100% Basemat)	2.44E-07	2.86E-09
Doubles every 2 yrs	Base	Base	2.47E-07	6.50E-09
Doubles every 10 yrs	Base	Base	2.43E-07	2.40E-09
Base	Base	15% Cylinder-Dome	2.45E-07	4.00E-09
Base	Base	5% Cylinder-Dome	2.43E-07	1.72E-09

Table 6.1-1
Steel Liner Corrosion Sensitivity Cases

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT EXTENSION FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base	10% Cylinder-Dome, 1% Basemat	Base	2.69E-07	2.86E-08
Base	0.1% Cylinder-Dome, 0.01% Basemat	Base	2.41E-07	2.86E-10
LOWER BOUND				
Doubles every 10 yrs	1.0% Cylinder-Dome, 0.1% Basemat	5% Cylinder-Dome 100% Basemat	2.41E-07	1.44E-10
UPPER BOUND				
Doubles every 2 yrs	10% Cylinder-Dome, 1% Basemat	15% Cylinder-Dome 100% Basemat	3.32E-07	9.10E-08

6.2 EPRI Expert Elicitation Sensitivity

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leaks within containment [22]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as a function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of failure mechanisms, the potential for undiscovered mechanisms, inaccessible areas of the containment as well as the potential for detection by alternate means. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage within containment. The base case methodology uses the Jeffrey's non-informative prior for the large leak size and the expert elicitation sensitivity study uses the results from the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the base case methodology (i.e., 10La for small and 100La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the base case and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, including the input to expert elicitation as well as the results of the expert elicitation, are available in the various appendices of EPRI TR-1018243 [22].

Table 6.2-1
EPRI Expert Elicitation Results

LEAKAGE SIZE (LA)	BASE CASE	EXPERT ELICITATION MEAN PROBABILITY OF OCCURRENCE [22]	PERCENT REDUCTION
10	9.2E-03	3.88E-03	58%
100	2.3E-03	2.47E-04	89%

The summary of results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the base case evaluation (10La for small and 100La for large). The expert elicitation process produces a relationship between probability and leakage magnitude in which it is possible to assess higher leakage magnitudes that are more reflective of large early releases; however, these evaluations are not performed in this particular study.

The net effect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall value for LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 2.58E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.08E-08/yr. As such, if the expert elicitation mean probabilities of occurrence are used instead of the non-informative prior estimates, the change in LERF for PLP is within the range of a "very small" change in risk when compared to the current 1-in-10, or baseline 3-in-10 year requirement. The results of this sensitivity study are judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the values from the EPRI methodology [22], and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

Table 6.2-2
PLP ILRT Cases:
3 in 10 (Base Case), 1 in 10, and 1 in 15 Yr intervals
(Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR
1	4.10E+04	8.90E-06	3.65E-01	8.65E-06	3.55E-01	8.47E-06	3.47E-01
2	6.15E+06	1.31E-07	8.06E-01	1.31E-07	8.06E-01	1.31E-07	8.06E-01
3a	4.10E+05	1.02E-07	4.16E-02	3.38E-07	1.39E-01	5.08E-07	2.08E-01
3b	4.10E+06	6.47E-09	2.65E-02	2.15E-08	8.83E-02	3.23E-08	1.33E-01
7 non-LERF	1.07E+06	1.29E-05	1.38E+01	1.29E-05	1.38E+01	1.29E-05	1.38E+01
7 LERF	6.15E+06	2.88E-07	1.77E+00	2.88E-07	1.77E+00	2.88E-07	1.77E+00
8 non-LERF	1.84E+06	4.28E-06	7.89E+00	4.28E-06	7.89E+00	4.28E-06	7.89E+00
8 LERF	6.15E+06	1.03E-13	6.36E-07	1.03E-13	6.36E-07	1.03E-13	6.36E-07
Total		2.66E-05	2.470E+01	2.66E-05	2.485E+01	2.66E-05	2.495E+01
ILRT Dose Rate from 3a and 3b		6.82E-02		2.27E-01		3.41E-01	
Delta Total Dose Rate[1]	From 3 yr	---		1.48E-01		2.55E-01	
	From 10 yr	---		---		1.06E-01	
3b Frequency (LERF)		6.47E-09		2.15E-08		3.23E-08	
Delta LERF	From 3 yr	---		1.50E-08		2.58E-08	
	From 10 yr	---		---		1.08E-08	
CCFP %		66.16%		66.22%		66.26%	
Delta CCFP %	From 3 yr	---		0.06%		0.10%	
	From 10 yr	---		---		0.04%	

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

7.0 CONCLUSIONS

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to fifteen years:

- Reg. Guide 1.174 [4] 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. "Small" changes in risk are defined as increases in CDF below 10^{-5}/yr and increases in LERF below 10^{-6}/yr. Since the ILRT extension was demonstrated to have a negligible impact on CDF for PLP, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included is 2.44E-07/yr (see Table 6.1-1), which falls within the small change region of the acceptance guidelines in Reg. Guide 1.174. In using the EPRI Expert Elicitation methodology, the change is estimated as 2.58E-08/yr (see Table 6.2-2), which falls within the very small change region.
- The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all accident sequences, is 1.34E+00 person-rem/yr using the EPRI guidance with the base case corrosion case (Table 5.6-1). The change in dose risk drops to 2.55E-01 person-rem/yr when using the EPRI Expert Elicitation methodology (Table 6.2-2). The value calculated per the EPRI guidance is slightly higher than the acceptance guideline for a "very small" change of ≤1.0 person-rem/yr defined in Section 1.3. However, this calculated increase is conservatively high based on the assignment of the L-LL release category to the intact containment case, which subsequently yields conservative estimates of the EPRI Class 3a and 3b calculated dose results. As such, the risk impact when compared to other severe accident risks is small.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance (see Section 5.5) is 0.91%, and drops to about 0.10% using the EPRI Expert Elicitation methodology (Table 6.2-2). Although no official acceptance criteria exist for this risk metric, it is judged to be very small.
- To determine the potential impact from external events, an additional bounding assessment from the risk associated with external events utilizing information from the PLP IPEEE was performed. As shown in Table 5.7-2, the total increase in LERF due to internal events and the bounding external events assessment is 4.9E-07/yr, which is in Region II of the Reg. Guide 1.174 acceptance guidelines.
- Finally, as shown in Table 5.7-3, the same bounding analysis indicates that the total LERF from both internal and external risks is 1.3E-06/yr, which is less than the Reg. Guide 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II (small change in risk).

Therefore, increasing the ILRT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be significant since it represents only a small change in the PLP risk profile.

Previous Assessments

The NRC in NUREG-1493 [5] has previously concluded the following:

- Reducing the frequency of Type A tests (ILRTs) from 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.
- 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 structure.

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

8.0 REFERENCES

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Appendix A
PRA Technical Adequacy

A.1 Overview

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this report to help support an extension of the Palisades containment Type A test integrated leak rate test (ILRT) interval to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 2 [1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed to perform this study:

1. Identify the parts of the PRA used to support the application
    - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model.
    - A definition of the acceptance criteria used for the application.
2. Identify the scope of risk contributors addressed by the PRA model
    - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
    - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
    - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
    - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
    - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.
    - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 are covered in the main body of this report. The purpose of this appendix is to address the requirements identified in item 4 above. Each of these items (plant changes not yet incorporated into the PRA model, relevant peer review findings, consistency with applicable PRA standards and the identification of key assumptions) are discussed in the following sections.

The risk assessment performed for the ILRT extension request is based on the current Level 1 PRA model analysis of record. Information developed for the license renewal effort to support the Level 2 release categories is also used in this analysis. Model updates have occurred and are discussed below.

None of these updates have significantly altered the CDF (core damage frequency) or the LERF (large early release frequency) values such that the bounding analyses performed herein are in question. For this application, the accepted methodology involves a bounding approach to estimate the change in the LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

To address the RG-1.200 requirements; however, Section A.2 provides a summary of the peer review results from past assessments of the Palisades PRA model. This also includes a report on the latest set of findings from the October 2009 peer review. An evaluation of the impact of these findings on the ILRT extension risk assessment is presented. Section A.3 provides an assessment of key assumptions and approximations used in this assessment and Section A.4 briefly summarizes the results of the PRA technical adequacy assessment with respect to this application.

A.2 PRA Model Review History

A.2.1 CEOG Peer Review

The CEOG conducted an industry peer review of the Palisades PRA in 2000 [2]. All level A and B findings have been addressed.

A.2.2 2004 Gap Analysis

Subsequent to the 2000 peer review a gap analysis was performed in 2004 [3].

At the behest of the NRC, the industry undertook a task to develop a consensus standard on the technical adequacy of PRAs for regulatory applications. This effort resulted in publication of ASME RA-S-2002. Concurrently, under the direction of the Nuclear Energy Institute (NEI) and the Owners Groups for each major reactor provider, peer reviews of PRAs were conducted using the guidance in NEI 00-02. The NRC was also concurrently developing guidance for determining the adequacy of risk analyses for use in regulatory applications. The first draft of this guidance was published as Draft Guide 1122 (DG-1122) in September 2002. Following interactions with industry in subsequent years as the ASME Standard was being modified, the NRC published DG-1161 in September 2006. This draft version of Regulatory Guide 1.200 (RG 1.200) provided guidance on self assessments to determine the adequacy of PRAs.

This assessment reviewed the peer review facts against the guidance in DG-1122 and produced a list of recommended actions to address "gaps" between the results of the peer review and the guidance in DG-1122. As noted above, Palisades had subsequently addressed all A and B level facts and observations (F&Os) from the peer review certification report. DG-1122 allowed for two mechanisms for conducting a self assessment. One was a direct comparison of the PRA against the Standard with additional considerations cited by the NRC to address areas where the NRC did not agree with the Standard (Table A-1 of DG-1122). The other method was to take advantage of the peer review findings and perform additional reviews against the Standard in areas where the NRC found that the peer review process needed additional effort to address NRC concerns with the Standard. The NRC issues were documented in Table B-4 of DG-1122. This was the method used in the Palisades Gap Analysis.

Table A.2.2-1 lists the recommended actions identified by this evaluation. In general, the additional recommendations deal with issues of documentation and/or justification for technical analyses in the PRA. Slightly less than half of the additional recommendations are likely to result in a change to the actual model. Only three additional recommendations are likely to result in a noticeable change in the CDF or LERF. These include the removal of EDG repair from the model, the inclusion of additional flow diversion paths for key systems, and the inclusion of potential concurrent unavailabilities (such as train wise maintenance schedules where one train in multiple systems is taken out of service at the same time.

The risk impact of the latter issue is bounded by the risk evaluations done to adhere to the a(4) requirements of the Maintenance Rule (10CFR50.69) and are not expected to be significant with respect to the baseline CDF evaluations.

Table A.2.2-1 Additional Recommendations to the Gap Analysis

Item	Description of Issues	Applicable SR Numbers	Model Changes Likely Needed?	Disposition
A	Document the rationale for not using "precursors" to identify initiators.	IE-A7	No	Addressed
B	Walkdowns/interviews with operators and engineers have been conducted in the past, but need to be done again in light of recent PRA updates and staffing changes.	SY-A4, IF-B3, IF-C8, IF-E8	No	Addressed
C	Flow diversions are included in many systems but additional cases need to be included in the model.	SY-A12b	Yes	Addressed
D	Concurrent unavailabilities should be included in the model.	DA-C13	Yes	Addressed
E	Palisades included inter-area propagation but needs to include unavailability of flood barriers such as doors/hatches.	IF-C3b	Yes	Addressed
F	Palisades credited flood isolation operator actions after 30 minutes. Further activity is underway to document the time available and the reliability of the potential actions.	IF-C7	No	Addressed
G	Generic and plant specific experience was used in determining pipe failure frequencies, but factors such as the impact of FAC, water hammer, etc. should be included in the analysis.	IF-D5a	Yes	Addressed
H	Key assumptions were documented but key uncertainties in the analysis need to be documented and evaluated.	IF-F3	No	Addressed

Table A.2.2-1 Additional Recommendations to the Gap Analysis

Item	Description of Issues	Applicable SR Numbers	Model Changes Likely Needed?	Disposition
I	ISLOCA evaluation included pressure capability of secondary systems. Capability for valve closure under high flow/dP to isolate ISLOCA was not credited. Document the rationale for this exclusion.	IE-C11	No	Addressed
J	The pre-initiators were identified primarily based on test and maintenance activities. Inspection activities also should be addressed explicitly for potential pre-initiators.	HR-A1	No	Addressed
K	The quality of procedures and processes were examined to the extent that the THERP methodology calls for, but do not include all the factors in the latest version of DG-1161. Document how the pre-initiator HEPs account for the quality factors noted in DG-1161.	HR-D3	No	Addressed
L	EDG repair is the only case where repair is credited. Palisades intends to remove that feature from the PRA.	DA-C14	Yes	Addressed
M	The flooding analysis did not consider ranges of flow rates for flood sources, but used maximum flow rates instead. Determine if lesser flow rates would impact the results and include as warranted.	IF-B3	Yes	Addressed
N	Barrier availability was generally not accounted for but reverse flow via failed check valves was included in the flooding analysis. Include potential barrier unavailability.	IF-C3b	Yes	Addressed
O	CCF groups were not reduced to account for the effects of flooding. This results in pessimistic (conservative) impact of CDF for flooding sequences. Document the rationale for not adjusting CCF group sizes for equipment that would be failed by flooding scenarios.	IF-E6a	No	Addressed
P	Sensitivity analyses on key assumptions have been performed over time but have not been documented in a comprehensive manner. Consider referencing sensitivity analyses in EA calculations in the documentation of the current version of the model and subsequent updates.	QU-E4	No	Addressed

A.2.3 October 2009 Full Power Internal Events Peer Review [4]

The final report documenting the results of full-scope Regulatory Guide (RG) 1.200 peer review for the Palisades Nuclear Power Plant Probabilistic Risk Assessment (PRA) was received on March 12, 2010.

The review concluded:

> *The Palisades PRA substantially meets the ASME PRA Standard at Capability Category II or better for 83% of the applicable Supporting Requirements, with 88% met at Capability Category I or better. This review resulted in eighty-one new F&Os, twenty-four "Suggestions," fifty-five "Findings" and two "Best Practices". As documented in AS-A9-01, the*

> *limitations of the T-H codes used are specifically documented in an Appendix in NB-PSA-ETSC, r.01, and all codes were used within their range of applicability. This is considered a best practice. Also, as described in SY-A13-02, the flow diversion pathways for CCS, CSS, LPSI, SWS, AFW, CVCS, and HPSI are performed systematically for all potential flow diversion paths and provide an excellent basis for the diversion paths modeled. This is also considered a best practice. However, there were several technical issues that should be addressed. These technical issues are summarized in section 4 with the details in Table 4-12.*
>
> *Overall, the Palisades PRA was found to substantially meet the ASME PRA Standard at Capability Category II and can be used to support risk-informed applications. Dependent upon the specifics of the application, additional supporting analyses may be needed, particularly for applications that impact elements with unresolved findings or where an assumption could impact the conclusions of the application.*

Most of the reported findings have been addressed. Fourteen of the internal flooding issues have been placed on hold as they are considered low priority issues. However, it should be noted that Palisades with support from Operations created a flooding Off Normal Procedure (ONP) with basis. This procedure was purposely written to collect and organize various flooding recoveries that were embedded in different Alarm Response Procedures. And the ONP basis document allows the forum for sharing the detailed GOTHIC deterministic results and insights gained from performance of the internal flooding analysis. The procedure is currently in review.

Most of the other F&Os as well as suggestions have been addressed. Refer to Table A.2.3-1 below. The table provides a paraphrased draft summary of the findings and resolutions. The fourteen internal flooding F&Os are not described. Several suggestions that were addressed are listed. The remaining suggestions not addressed are not listed in the table below. However, Palisades still met category II with respect to the "not" listed supporting requirement's suggestions.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
AS-A2-01	Finding	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage.	The event trees specify the required key safety functions needed to mitigate the initiating event of interest, but the mission time is specified as 24 hours. Need to ensure all end states are safe, stable states at the 24 hour mission time, or extend the mission time until a safe, stable end state is met for each accident sequence. There is not sufficient documentation that 24 hours is appropriate to ensure that all accident sequences reach a safe, stable end state. Also, not all end states specified on the event trees may be correct.	Notebook NB-PSA-SS, "Palisades Safe and Stable States" [23] was developed to evaluate and document the non-core damage end states for all event trees. Generalized flow charts were developed to capture all of the non-core damage sequences based on the general transient/main steam line break, loss of offsite power, loss of cooling accident, very small break loss of coolant accident (consequential LOCA) and steam generator tube rupture event trees. Event tree headings were translated in the flow charts to decision boxes allowing a path to be followed to reach the "OK" end states. ILRT analysis – no impact.
AS-A3-01	Finding	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A3), IDENTIFY the systems that can be used to mitigate the initiator.	The documentation associated with the event trees does not always match the current event tree logic. For some of the event tree nodes there appears to be a documentation mis-match. For example: Section 5.9 of NP-PSA-ETSC, r01 states that 3 of 3 charging pumps are required for a VSBLOCA, but the success criteria for the event tree top logic (CHRG-FT) states the success criteria is 2 of 3 charging pumps.	The success criteria notebook, NB-PSA-ETSC [11] was revised to ensure all event tree headings match the headings described in the notebook documentation. Section 4.9 was added to the notebook to describe the operation of auxiliary feedwater pump P-8B after battery depletion heading. Section 5.9 was corrected to agree with the number of required charging pumps as described in the success criteria in Section 5.9.4 for the CHRG-FT event tree heading. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
AS-A10-01	Finding	In constructing the accident sequence models, INCLUDE, for each modeled initiating event, sufficient detail that significant differences in requirements on systems and required operator interactions (e.g., systems initiations or valve alignment) are captured. Where diverse systems and/or operator actions provide a similar function, if choosing one over another changes the requirements for operator intervention or the need for other systems, MODEL each separately.	Although the event trees include operator actions required for success of key safety functions, the documented actions do not include verification that the operator actions, as evaluated, are "bounding" for all event tree nodes where the operator action is applied. The CAT II requirement to capture and provide sufficient detail for significant differences in requirements associated with systems and/or operator responses is not performed. For example, the event tree node PORV-FT appears in multiple event trees including Main Steam Line Break (MSLB), SGTR, LOBUS1A, PCP-SBLOCA, LOOP, but the Operator action is based on timing for Loss of Main Feedwater (LOMFW). There is no differentiation between the timing for any of the other initiators, and it does not appear that the LOFW initiating event is the bounding event for this operator action. The operator actions as currently evaluated need to be reviewed to ensure they are "bounding" for all scenarios where they are credited. To meet the CAT II requirement, timing differences (and potentially stress levels, etc.) need to be addressed for each accident sequence where the operator actions are credited.	Validation of nearly all the full power internal event HEP analysis is nearly complete. The validation documentation is included in [8]. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
AS-C2-01	Finding	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results events); (d) the operator actions reflected in the event trees, and the sequence-specific timing and dependencies that are traceable to the HRA for these actions; (e) the interface of the accident sequence models with plant damage states; (f) [when sequences are modeled using a single top event fault tree] the manner in which the requirements for accident sequence analysis have been satisfied.	There are some event trees that are not well documented in the Accident Sequence or Initiating Event notebooks. No documentation associated with the success criteria could be found for the Controlled Manual Shutdown Event tree. Additionally, there are multiple additional event trees (referred to as "Special Initiators") in the SAPHIRE program that are not explicitly discussed within the accident sequence documentation. A discussion of how an FMEA was performed to identify plant-specific system initiators is included in the Initiating Event notebook, however, the FMEA provided in the report is an "example FMEA," but the actual FMEA performed is not included or referenced. No discussion could be found that identifies how the final support system initiators were identified, how they are grouped, or how the event tree branches were defined. Since these event trees appear to use the same branches as other event tree, their "grouping" needs to be discussed, including the appropriateness of using the same event tree nodes for the event trees. Without a discussion of the event trees and nodes associated with the support system initiators, there is no documentation that the key safety functions or success criteria defined is appropriate and adequate for them. Although the Controlled Manual Shutdown is listed in the table in Attachment 3, there is no mention of it in the discussion in Section 3. A discussion of the event needs to be included in Section 3 similar to how the other transients are described. Also need to include the actual FMEA in the documentation or provide a valid reference for it.	Added Table 3.0-1 and supporting discussion to Section 3.0 of the event tree and success criteria notebook NB-PSA-ETSC to clarify that all transient initiators, including 'controlled manual shutdown' are applicable to the general transient event tree and its associated event tree headings and success criteria. The table and associated discussion also describes logical operators that are set to 'True' in the event tree rules file to establish the appropriate boundary conditions for each support system transient event. This discussion justifies the grouping of these initiators as applicable to the transient event tree. The result of the completed FMEA for all Palisades systems was developed into the 'Support System to Front-Line System Dependency Matrix' and 'Support System to Support System Dependency Matrix'. This was clarified in Section 2.2 of the initiating event notebook, NB-PSA-IE. The final FMEA results were added as Attachment 12 to the initiating events notebook. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
DA-A2-01	Finding	ESTABLISH definitions of SSC boundaries, failure modes, and success criteria in a manner consistent with corresponding basic event definitions in Systems Analysis (SY-A5, SY-A7, SY-A8, SY-A9 through SY-A14 and SY-B4) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events in a manner consistent with corresponding definitions in Systems Analysis (SY-A19).	Component boundaries defined for some Palisades components are not consistent with the generic data component boundaries for the same component. For example, the Palisades data report states that the generic data for motor-driven pumps includes the pump breaker, while the corresponding Palisades component boundary separates the pump breaker and pump into two separate events, with separate failure rates for each. This separation also appears to propagate to the definition of component boundaries for common cause failures. Component boundaries need to be consistent to avoid potentially double counting failures. Keep the separate basic events in the model, but assign a failure probability of "0" to the breaker and assign the "total" failure rate to the pump itself - including updating the generic data with the "total" plant-specific failures (pump and associated breaker failures), and calculating the corresponding CCF data based on the total failure rate. This allows sensitivities and insights to be obtained using the circuit breakers, while ensuring the model meets the component boundary requirements of the standard. If differences between component boundaries defined in the Palisades PRA and those in generic databases are retained, these differences and their bases should be included in the PRA documentation.	The Palisades PRA modeling intentionally separates contact pairs, breakers etc. from pump motors. This is the correct method of modeling plant equipment to ensure that appropriate qualitative insights are realized. This practice was demonstrated during conduct of the Industry IREP initiative in 1980 and 1981. Moreover, this practice was adopted in the development of the Palisades logic modeling that commenced in 1982 in support of the MSIV SEP issue resolution. In addition, an evaluation was performed using an interim model (PSAR3 Release 2b) to determine the magnitude of the potential conservatism introduced by having separate data and component boundaries for breaker-pump combinations as well as other components supplied with electrical power via breakers. To bound the problem described in finding DA-A2-01, a change set was developed with the failure probability for all breakers in the PRA (125 dc, 125 ac, 480 ac, 2400 ac, and 4160 ac) set to zero. The release 2b base model core damage frequency with the normally applied breaker failure probability was 2.29 E-05/yr. With the failure probability of all breakers set to zero, the core damage frequency reduces 19% to 1.85 E-05/yr. This change is less than a factor of two different and is essentially the same result. As this difference in total CDF is considered negligible, and the additional resolution and insights gained are far more valuable than the demonstrated conservatism, it is deemed unnecessary to redefine component boundaries in the PRA. In summary, modeling equipment subcomponents such as breakers, relays, contact pairs, hand switches etc. ensures a comprehensive qualitative characterization of systems, structures and component reliability. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
DA-C7-01	Finding	BASE number of surveillance tests on plant surveillance requirements and actual practice. BASE number of planned maintenance activities on plant maintenance plans and actual practice. BASE number of unplanned maintenance acts on actual plant experience.	Palisades used actual plant procedures and experience to count surveillance tests. Planned maintenance activities are estimated rather than being based on maintenance plans. Contribution from planned maintenance is based on previous operating experience and not based on maintenance plans which might be different from the previous plant experience. Review planned maintenance activity plans or review existing estimates with Maintenance personnel to determine whether estimates of planned maintenance should be changed.	All preventive maintenance activities for PRA identified components were collected from Palisades' current equipment database and reviewed. Referring to sections 5.4 of the Data notebook, NB-PSA-DA, Palisades pm data is based on active, planned PMs. The scope of this project was three calendar years of plant operation. PM data was actually counted for preventive maintenance tasks (PMs) with frequencies of three years or less. For active PMs with frequencies greater than three years, "an equivalency was defined based on the three year data window". For example, if there was a PM that occurred every 6 years, a PM frequency of 0.5 was assigned. While this number is estimated, it is based on the number of actual active, planned PMs. This approach to modeling was reviewed with a qualified PM Program Engineer and validated that this approach represents realistic representation of PM frequency. Documentation of the review was added to Section 5.4 [15]. ILRT analysis – no impact.
DA-D1-01	Finding	CALCULATE realistic parameter estimates for significant basic events based on relevant generic and plant-specific evidence unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. When it is necessary to combine evidence from generic and plant-specific data, USE a Bayes update process or equivalent statistical process that assigns appropriate weight to the statistical significance of the generic and plant-specific evidence and provides an appropriate characterization of uncertainty. CHOOSE prior distributions as either noninformative, or representative of variability in industry data.	Bayesian updates of all plant specific calculations used the industry average distributions. For Category II, it is necessary to update significant basic events using a non-informative prior or a prior that represents the variability in industry data. For significant components, the use of the industry average prior may have distribution spreads that can overwhelm plant experience data when doing a Bayesian update. Use of the a constrained non-informative prior or a prior reflecting plant to plant variability would allow plant operating experience to have a larger impact on the resulting posterior mean. Review the significant basic events and evaluate the plant specific updates based on a constrained non-informative prior or a prior based plant variability.	Palisades parameter estimates are calculated based on Bayesian analysis employed with a combination of plant specific and generic industry sources. Generic industry sources include NUREG/CR-6928, NUREG-1715 Volume 4, EPRI TR-016780 Rev. 6, NUCLARR, NUREG/CR-4639, ASEP, and NUREG/CR-4550, as documented in Attachment 10 of NB-PSA-DA [5]. Plant specific data sources of failure data included a review of some 10,000 plant work orders and review of documented maintenance rule failures as listed in Attachment 3 [5]. Prior distributions were selected to represent variability in the industry data when generic sources were applied. The Bayesian update process was performed using the BART code which graphically illustrates the prior and posterior distributions on the same plot. During this process, there were no instances during the update of important basic events where it was observed the generic industry data overwhelmed the plant specific data resulting in a posterior that had very little or no change relative to the prior. Therefore, the use of generic industry distributions in lieu of a non-informed prior for significant basic events was appropriate. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
DA-D4-01	Finding	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Where generic data was Bayes-updated with plant-specific data, self-checks should be performed and documented to ensure that the posterior distribution was reasonable. Based updated data should be confirmed appropriate. It is suggested that the data notebook include a discussion of how the requirements of this SR DA-D4 are met.	All Bayesian update results were reviewed. In cases where there were no plant failures, demand results for means below 1E-06 and run time rates below 5E-06 were reviewed to ensure they were not unrealistically low. In all cases, changes in the mean were negligible (i.e., less than a factor of 2). In cases where there were no plant failures and the failure rates were above 1E-06 for demands and 5E-06 for run times, the results were reviewed to confirm the impact from the Bayesian update was minimal (i.e., less than a factor of 3). Attachment 11 [6] provides a comparison of the posterior mean next to the prior. In addition, a comparison was made between the data used in the previous analysis [7] to that used in this update. Failure codes in which there was a measurable difference in the Bayesian updated plant-specific data (e.g., factor greater than 5) were reviewed in detail. A spreadsheet analysis for each Bayesian update was performed using the BART code which provides a visual comparison of the prior and posterior distributions. The Bayesian update process and reviews performed are described in Section 8.1 of NB-PSA-DA [5]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
DA-D8-01	Finding	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: (a) If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available for significant basic events; or (b) If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and assess the hypothetical effect on the historical data to determine to what extent the data can be used.	Plant-specific failure data collected in the collection data window must be poolable and applicable to the current plant. Only applicable plant-specific data can be applied to the failure events. In order to ensure that plant-specific data collected in the collection data window is poolable and applicable to the current plant, plant modifications (both hardware and procedural) implemented during this time period should be reviewed for potential impact for on this failure data. This review should be documented and the use of plant-specific data should be limited, as appropriate.	Section 4.3, "Plant Modifications," was added to the Palisades PSA Data Notebook, NB-PSA-DA rev. 5 [5]. This section of the notebook documents that a review of plant modifications during the data window was performed. A complete list of modifications performed during this time was added to Attachment 3 of the document. ILRT analysis – no impact.
HR-A1-01	Finding	For equipment modeled in the PRA, IDENTIFY, through a review of procedures and practices, those test and maintenance activities that require realignment of equipment outside its normal operational or standby status.	Identification of Pre-Initiator HFEs- No pre-initiator HFAs are included for the AFW pump train restoration, common AFW suction from the CST, EDG restoration, High Pressure Safety Injection (HPSI) pump train restoration, Low Pressure Safety Injection (LPSI) pump train restoration, etc. No documentation was provided on the decision making process for excluding restoration errors for standby components such as these. Restoration of pump train for standby systems can be a contributor to risk. Review each system for possible pre-accident restoration errors and if such events are not included in the model, provided a basis for exclusion. The process identified for screening pre-initiator human failure events should be sufficient to identify most pre-accident HRAs.	The pre-initiator process was revised to include a process of assessing each system. The initial step of the HFE identification process was to identify the plant systems to be considered in the review. The Palisades pre-initiator methodology [8] indicates that the review should include all systems modeled in the PRA, which are listed in the Palisades System Notebooks. Once the initial systems list was assembled, the system descriptions and simplified P&IDs were examined to identify and define the Train/Function/Channel (TFC) for the system. Those TFCs not susceptible to Type A (pre-initiator) events were screened from further review (this process is documented in Reference 8). For each of the unscreened TFCs identified, a scoping event was added to the PRA model. The scoping values were then used to determine the risk significance of each event and evaluate which events should remain in the model. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
HR-A2-011	Finding	IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.	Miscalibration events for the containment pressure instruments are missing without a detailed screening. This is similar to the F&O for the restoration events except for the calibration events. The miscalibration events appear to be more complete than the restoration events but additional work is necessary to identify the potential miscalibration events. Miscalibration of containment pressure signals would impact auto start of HPSI, LPSI, and Containment Spray System (CSS). It might also impact auto start of containment unit coolers and CIS signals. Review each system for possible pre-accident restoration errors and if such events are not included in the model, provided a basis for exclusion. The process identified for screening pre-accident human actions should be sufficient to identify most pre-accident HRAs.	The pre-Initiator process was revised to include a process of assessing each system. This system level review included potential miscalibration events, including those for the HPSI, LPSI, and containment spray system described in this finding. The initial step of the HFE identification process was to identify the plant systems to be considered in the review. The Palisades pre-initiator methodology [8] indicates that the review should include all systems modeled in the PRA, which are listed in the Palisades System Notebooks. Once the initial systems list was assembled, the system descriptions and simplified P&IDs were examined to identify and define the Train/Function/Channel (TFC) for the system. Those TFCs not susceptible to Type A (pre-initiator) events were screened from further review. For each of the unscreened TFCs identified, a scoping event was added to the PRA model. The scoping values were then used to determine the risk significance of each event and evaluate which events should remain in the model. ILRT analysis – no impact.
HR-C2-01	Finding	INCLUDE those modes of unavailability that, following completion of each unscreened activity, result from failure to restore (a) equipment to the desired standby or operational status (b) initiation signal or set point for equipment start-up or realignment (c) automatic realignment or power ADD failure modes identified during the collection of plant-specific or applicable generic operating experience that leave equipment unavailable for response in accident sequences.	Pre-initiator human failure events were included in the fault tree at the appropriate level for the pre-initiator HFE identified. However, based on the missing HFEs identified in F&Os against HR-A1 and HR-A2, and no evidence of a review of plant specific mispositioning or miscalibration events, credit cannot be given for collection of plant-specific or generic operating experience. No review of plant misposition or miscalibration and missing events generally included for standby components and instrumentation as discussed in HR-A1 and HR-A2. Perform a systematic review of HFEs. Consider a Condition Report review of mispositioned or miscalibrated events to determine if any trends associated with the pre-accident events could impact the HRA values.	The pre-initiator methodology was revised and each system re-evaluated for the possibility that pre-initiator events could occur at the train/channel/function level of each system. The revised methodology and new pre-initiator HEPs are discussed in the HRA Notebook Volume II (NB-PSA-HR [8]. Those pre-initiators specifically identified during the review and several others were assessed using the revised methodology and, as necessary, were added to the model. A review of plant history was conducted for plant specific operating experience. The result of the review was that while there were instances noted of conditions that would be considered pre-initiators, the examples noted were either already covered by a pre-initiator event identified during the implementation of the revised methodology or were related to equipment not credited in the PRA. The plant operating experience review is documented in HRA notebook volume 1 [25]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
HR-E3-01	Finding	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures.	HRAs were reviewed by former SRO to ensure and confirm that interpretation of the procedures is consistent with plant observations and training procedures. No review by training personnel was performed as required by Cat II & III. No review by training personnel was performed as required by Cat II & III. Document the talk through performed with training personnel to confirm the interpretation of the procedures is consistent with plant observations and training procedures.	A copy of the Human Error Probability (HEP) Post-Initiator Calculations (P-IC) and associated Post-Initiator Operator Action Questionnaire (P-IOAQ) were provided to current SRO licensed on-shift Operations Department personnel and Training Department personnel for use in validating HEP information accuracy. HEPs were assigned to the five Operations Department Operating Crews (~10 per crew) for review. Their reviews included ensuring indications, procedure selection and use, and activity performance man-power and timing is correct. Training personnel reviews included ensuring procedure selection and use were consistent with current training expectations, and the training type and frequency are accurate. Operator comments were reviewed and proposed resolutions forwarded to the comment initiator for further comment or acceptance. Comment initiator acceptance is documented by their initialing the HEP Validation form. The records of the current operating crews and training personnel are provided in Attachment F [25]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
HR-G6-01	Finding	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	HRA Procedure 5.3.2.12 states: "The consistency of resulting post-initiator Human Error Probabilities (HEPs) should be checked: (a) REVIEW the Human Failure Events (HFEs) and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience. (b) One approach for checking the consistency of HEP quantifications is to sort by increasing or decreasing HEP values and then performing the comparison." In addition, HRA Notebook Section 4.0 states: "After the individual results were obtained, the final HEPs were assessed for appropriateness and consistency within the PLP HRA. Human action element such as time frame and complexity of the action's diagnosis and/or execution were considered. When available, HEP results for similar actions at other PWRs were used as further points of reference." However, no documentation of the review was found. Consistency check is required for Capability Category I, II, and III. Document the consistency check that was performed.	A comparison of the human error probabilities (HEPs) developed for each human failure event (HFE) in the PLP internal events PRA model shows that the values of the HEPs are internally consistent relative to each other, and generally follow a trend of lower HEPs being associated with lower stress levels (which in turn may be associated with more time available to take action). Exceptions to the general trend are present, and can be explained through detailed examinations of the contributions to the HEP (e.g., number of procedure steps, time available to perform the steps, probability of successfully recovering from errors occurring during completion of the procedure, etc.). This review is documented in NB-PSA-HR Volume 1 [25]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
HR-G7-01	Finding	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) the time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel)	Palisades has not completed their HFE Dependency Evaluation for their updated HRA. This is specifically noted in Section 5.2 of PLP-HRA. Failure to meet explicit requirement of the standard. After the HRA is complete, redo and document the dependency evaluation.	The detailed methodology for evaluating human error dependency was completed as described in HRA Notebook NB-PSA-HR Volume 1, Section 5.2 [25]. However, the results for the FPIE analysis are not finished. This analysis evaluates the dependency between the multiple operator actions that occur in the accident sequences of the Palisades PSA. The human reliability analysis of the PSA developed human error probabilities (HEPs) as though they were independent of one another. It is known that a number of these operator actions appear in the same accident sequences. If dependencies exist between these operator actions, then the core damage frequency may be higher than quantified in the accident sequence analysis. This analysis evaluates the post-initiator dependencies among operator actions credited in the Palisades PSA and determines whether the impact of these dependencies on the overall core damage frequency is significant. The most risk significant human error dependencies were fully developed into conditional human actions and incorporated explicitly in the Palisades PSA fault trees. The general steps used in this analysis were as follows: 1.Run the base model with the post-initiator action failure event probabilities set to 1.0. 2.Identify the multiple human action combinations that appear in the cut sets. 3.Identify the risk significant combinations assuming complete dependence. 4.Perform a dependency analysis on the risk significant combinations and develop conditional probabilities for dependent actions. 5. Incorporate the dependent combinations in the fault trees of the PSA. To address the human action dependency issue with respect to CDF, Palisades developed a systematic approach that investigated a sufficient number of human actions to merit confidence that the impact of these dependencies have been thoroughly assessed and adequately represented in the PSA models. The approach is iterative and methodical. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
HR-I3-01	Finding	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the human reliability analysis.	There are only two assumptions in the entire HRA notebook. Both are associated with individual HRAs. General assumptions associated with HRA minimum defaults and methodology requirements are not listed as assumptions and are thus not addressed in terms of model uncertainty. Only two assumptions were listed for all of the HRAs. This does not appear to be consistent with the remainder of the model in terms of assumptions and sources of model uncertainty. Review the HRA for additional imbedded assumptions and use the updated list for potential model uncertainties.	Table 1.6.1 was added to the Human Reliability Analysis Notebook NB-PSA-HR Volume 1 [25]. This table documents some 65 assumptions including basis, assumption type, and model uncertainty impact. The assumptions are categorized into fire related and general HRA methods assumptions. ILRT analysis – no impact.
IE-A6-01	Finding	When performing the systematic evaluation required in IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	Although it appears that an evaluation of CCFs was performed since IE_LOY10-Y20 and IE_LO¬ALL4PREFAC, etc, were identified; however, documentation of the systematic evaluation for the elimination of other support system CCF events was not provided. Provide documentation of the evaluation of electrical equipment CCF initiating events.	A detailed evaluation to address this finding was completed in Attachment 3 [9]. In summary, the evaluation states: A process to ensure all possible common cause combinations and routine and non-routine system alignments is theoretically achievable, but time consuming and inconsistent with risk-informed approaches as utilized in PRAs. However, a process to ensure that combinations of failure events and system configurations that have occurred or could reasonably occur is in place already through the current Palisades approach of considering plant and generic data, initiating event categorization, and technical specifications. This process addresses reasonable common cause combinations if in fact such combinations are necessary to result in a plant trip. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
IE-A6-02	Finding	When performing the systematic evaluation required in IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	Event trees for common cause failures (e.g., Loss of Preferred AC Bus Y20, Y30, and Y40) are included in the SAPHIRE program, but no documentation associated with these event trees has been found. Note, the FMEA discussion provided in the Initiating Event notebook, does not specifically discuss the CCF initiators, nor does it identify the buses as necessarily resulting in a reactor trip. No discussion of non-routine system alignments has been found. The NRC's clarification for this element requires consideration, and documentation of Initiating events resulting from common cause or from both routine and non-routine system alignments. A systematic approach to ensure all possible common cause combinations and routine and non-routine system alignments needs to be developed, and documented.	A detailed evaluation to address this finding was completed in Attachment 3 [9]. In summary, the evaluation states: A process to ensure all possible common cause combinations and routine and non-routine system alignments is theoretically achievable, but time consuming and inconsistent with risk-informed approaches as utilized in PRAs. However, a process to ensure that combinations of failure events and system configurations that have occurred or could reasonably occur is in place already through the current Palisades approach of considering plant and generic data, initiating event categorization, and technical specifications. This process addresses reasonable common cause combinations if in fact such combinations are necessary to result in a plant trip. Recognize that given the plant's asymmetries the CCF grouping is straightforward. These results indicate that the impact of CCF (e.g., the preferred ac buses) is minimal when considering random failures. The PRA is satisfactory to achieve at least a Category II compliance with the ASME Standards IE A5 and A6. ILRT analysis – no impact.
IE-A8-01	Finding	INTERVIEW plant personal (e.g., operations, maintenance, engineering, and safety analysis) to determine if potential initiating events have been overlooked.	No meeting minutes or documentation of reviews performed by Licensed operators, system engineers and maintenance and training staff members to ensure that no potential initiating events have been overlooked. Lack of documentation of reviews performed by Licensed operators, system engineers and maintenance and training staff members to ensure that potential initiating events have been overlooked. Document review of IE List for comprehensiveness performed by Licensed operators, system engineers and maintenance and training staff members	Section 2.2 of NB-PSA-IE [10] was revised to document interviews and reviews of the PRA initiating events by specific plant personnel including the Assistant Operations Training Manager, Maintenance Rule Program owner, and two operations personnel. In addition, the current Palisades PRA personnel also act as the site safety analysis (Chapter 14) calculation owners. Interviews with System Engineers were performed by the PRA personnel and documented in Attachment 5 of all PRA system notebooks. These interviews included discussion of initiating events. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
IE-A9-01	Finding	REVIEW plant-specific and review industry operating experience for initiating event precursors, for identifying additional initiating events. For example, plant-specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	Evaluation of precursors mentioned in Section 2.2.6 Special Initiators as "Special initiating events or the potential for such events (e.g., precursors) was performed during the PRA teams' review of the Maintenance Rule (MR) database and Maintenance Work Orders (MWO) in support of the data effort." However, documentation of the specific review for precursors was not provided. Provide documentation to show the evaluation performed.	A documented review of all maintenance rule and work order failures was added to Section 2.2.6 of the initiating events notebook NB-PSA-IE [10] to determine if they are potential precursor events. Component failures were obtained from Attachment 3 of the data notebook NB-PSA-DA [5] and individually evaluated as to their potential as a precursor event. No new initiating events were developed as a result of the evaluation. However, the exercise did confirm several existing transient initiator events were appropriately modeled in the PRA. ILRT analysis – no impact.
IE-C2-01	Finding	When using plant-specific data, USE the most recent applicable data to quantify the initiating event frequencies. JUSTIFY excluded data that is not considered to be either recent or applicable (e.g., provide evidence via design or operational change that the data are no longer applicable.)	Justification for the exclusion of data before January 2003 used to identify plant-specific initiating events was not provided. Justification for the exclusion of data before January 2003 used to identify plant-specific initiating events was not provided. Provide the requested justification.	Added additional justification for the exclusion of data prior to January 2003 to Section 4.1 of the initiating events notebook NB-PSA-IE [10]. Justification is based on improved plant availability from January 2003 – 2009 relative to the previous site specific initiating event data from January 1994 – December 2002. Improvements in plant availability were demonstrated graphically in Figure 4.1. Plant availability has demonstrably improved after January 2003 due to improved operating and maintenance practices. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
IE-C6-01	Finding	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events or groups from further evaluation: (a) the frequency of the event is less than 1E-7 per reactor year (/ry), and the event does not involve either an ISLOCA, containment bypass, or reactor pressure vessel rupture (b) the frequency of the event is less than 1E-6/ry, and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator, or (c) the resulting reactor shutdown is not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically). If either criterion (a) or (b) above is used, then CONFIRM that the value specified in the criterion meets the applicable requirements in Data Analysis (2-2.6) and Level 1 Quantification (2-2.7).	In relation to IE-C6, Operator actions are apparently credited for the exclusion of some events (e.g., CRHVAC refer to earlier HVAC comments) without justifying each such credit (operator training, procedures, etc.) If component/system failures lead to an initiating event but are screened from further analysis by crediting operator actions or equipment/systems to avert the transient, then quantify the total initiating event frequency considering these events and apply criteria of IE-C6 to determine if screening criteria is met. Apply IE-C6 screening criteria and document as appropriate.	The basis for excluding control room HVAC from the full power internal events model was strengthened to include other aspects in addition to operator actions and was fully documented in Attachment 8 of NB-PSA-ETSC [11]. The evaluation was updated to include discussion of the control room heat-up rate effects on the reactor protective system (RPS) components and concluded that a loss of HVAC would not result in a significant increase in the failure probability of the RPS. In addition, a comparison of sensitivity analyses performed based on 14 owner's group sites that modeled the contribution to CDF due to loss of control room HVAC. The sensitivity studies found that the average CDF/yr was 1.61E-07 with a median of 1.31E-07/yr. Given Palisades core damage frequency is on the order of E-05, the change in CDF due to loss of control room HVAC would be less than 1%. With respect to cable spreading room cooling. An analysis of the cable spreading room heat-up following a loss of ventilation was developed using the GOTHIC software code and documented in EA-PSA-GOTHIC-CSRHEATUP-09-09 Rev. 0 [12]. This analysis developed a conservative room heat-up profile based on actual test data and assuming operators take no action to either open doors or affix portable ventilation. Using the room heat-up profile output from the analysis, CALC-455-001-DC2 [13] was then performed to evaluate all cable spreading room equipment modeled in the PRA at the predicted peak temperature for 48 hours. Based on the evaluation of equipment qualification reports, and vendor data, it was concluded there is reasonable assurance of operability for all equipment in the room under these conditions. The conclusions of these analyses demonstrate ventilation to the cable spreading and control room areas is not necessary to be explicitly modeled and the bases for these conclusions do not require operator action to mitigate elevated temperatures. However, ventilation is considered for purposes of fire modeling in these areas. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
LE-C9-01	Finding	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	No credit is taken for equipment operability or operator actions for adverse environment or containment failure. In Section 6.2.4 of the Level 2 report, Palisades stated that they had reviewed the results for cases where credit for equipment or HRAs during harsh environment or after containment failure might be applicable but did not justify equipment survivability in either of these conditions based on the contention that there were no cases where crediting continued equipment operation or operator actions would affect LERF. Therefore no credit was taken for continued equipment operation or operator actions. This clearly meets the requirements for Capability Category I. To move up to Capability Category II/III, i.e., getting credit for not crediting equipment, Palisades would need to provide much more documentation on what was looked at for equipment operability or operator actions and provide the bases for why the equipment would not be operable or that crediting the equipment made no difference to LERF. This should be tied to the Severe Accident Mitigation Guidelines (SAMG).	No credit is taken for equipment operability or operator actions in adverse environments or after containment failure. Palisades reviewed the LERF results for opportunities to take such credit (as documented in Section 6.2.4 of the Level 2 Notebook) and justified the lack of credit. Based on way the standard is written, the only way to earn a CC-II categorization is to credit equipment operation in adverse environment (for LE-C9 and C-10) and after containment failure (for LE-C11 and C12). Moreover, from an equipment context, Palisades does credit equipment in containment in environments that are considered beyond the EEQ harsh environment for which the equipment is qualified in the design basis. The MAAP program was utilized in calculation PLP0247-07-0004.01R0 [14] to determine the bounding best-estimate containment environmental conditions postulated to be encountered by equipment located in containment and modeled in the PRA. Both single and double steam generator blowdowns inside containment as well as once-through-cooling events were analyzed, with either a single containment air cooler or a single containment spray pump and spray header available. Additional variations with respect to steam generator isolation and auxiliary feedwater flow were analyzed. The limiting conditions are considered to represent the worst containment conditions expected prior to core damage and vessel failure, and are clearly beyond the design basis of the plant given the assumption of a double steam generator blowdown and that only portions of redundant containment heat removal systems are available. Calculation CALC-455-001-DC1 [15] evaluates the survivability of equipment modeled in the PRA under the environmental conditions determined in the MAAP analyses. This analysis utilized temperature profiles from the MAAP program to demonstrate that all credited PRA equipment located in containment can survive the limiting containment conditions produced by MSLB, LOCA, and OTC scenarios in which only a single containment air cooler or a single containment spray pump and header are available. A further detailed summary is provided in Attachment 8 of NB-PSA-ETSC [11]. In summary, it is considered that supporting requirements LE-C9 and C10 meet the CC-II requirements as the above noted engineering evaluations provide the justification. However, as no equipment has been explicitly shown to be available after containment failure, a CC-I classification of supporting requirements LE-C11 and C12 is considered appropriate. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
LE-G5-01	Finding	IDENTIFY limitations in the LERF analysis that would impact applications.	The Palisades PSA Level 2 Notebook does not explicitly discuss any limitations in the LERF analysis that might impact applications. It is expected that the limitations will be similar to those discussed for the level 1 analyses, but the level 1 discussion does not explicitly cover LERF so their analysis does not comply with the SR. Palisades should develop such a discussion similar to that developed for the level 1 analyses or revise the level 1 discussion to include LERF.	Given the Palisades two source term models, PAL-L2 and PWROG-L2, it is considered that sufficient detail exists such that this requirement is met. However, consideration of developing guidance will be provided and the Level 2 notebook will be updated [26]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
MU-B2-01	Finding	Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are incorporated as soon as practical.	The Palisades analysis of record is PSAR2c. The model was revised to include modifications needed for the internal flooding analysis. The current release presented to the peer review team is PSAR3 Release 2b. This model contains the updates associated with the requirements of Reg Guide 1.200 as well as changes to address NFPA-805. PSAR3 Release 2b is not the current analysis of record.	The Palisades analysis of record is PSAR2c . The version release presented to the peer review team on 10/26/10 was "PSAR3 Release 2b". The PSAR3 release series is a set of updates that address Reg. Guide 1.200, as well as a variety of NFPA-805 issues from MSO's to spurious CHP. These releases are not validated analyses-of-record models yet. The purpose of providing the PSAR3 Release 2b results to the peer review team was to show the latest consequences from a variety of model updates ranging from component random failure data, IE frequencies etc., updating to addressing extensive flow diversion scenarios, to adaptation of the simplified Westinghouse LERF model, to incorporation of a new comprehensive common cause model that employed the latest data, etc. PSAR3 Release 2b differed from PSAR3 Release 2a due to inclusion of new IE, HRA data etc. With exception of the last significant plant modification (GSI-191 sump strainers) that was finalized in the spring of 2009, all significant modifications had been addressed in the current analysis-of-record, the PSAR2c model dated 6-30-2006. Both PSAR3 Release 2# series included the finalized GSI-191 modifications. These GSI-191 physical modifications were completed in the spring of 2009 during the scheduled REFOUT. These modifications included an extensive re-analysis of the passive screen design due to the reconciled chemical-effect tests that occurred in the fall of 2008. Given the PRA teams design basis responsibilities, support of this initiative was very involved and included support of an onsite NRC inspection that was required for startup approval. The GSI-191 mod was the most significant plant change that had occurred since release of the current analysis-of-record dated 6/30/2006 (PSAR2c). Per completion of Release 2c, the Palisades QA process etc. will be applied in establishing a "new analysis-of-record" which will be termed PSAR3 Fire [16]. This model will support the NFPA-805 initiative. The PRA model is a living analysis. The configuration management procedures are applied to control, develop, and adapt parallel models. This is not considered a finding. ILRT analysis – no impact. The GSI-191 modification addressed the uncertainty associated with LOCA generated debris and its impact on the plants recirculation actuation system. The reliability of the original plant sump strainers is considered to have improved given the addition of several orders of magnitude of additional screen surface area.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
MU-B4-01	Finding	PRA Upgrades shall receive a peer review (in accordance with the requirements specified in Section 6 of the ASME PRA Standard) for those aspects of the PRA that have been upgraded. Refer to Section 2 of the ASME PRA Standard for the distinction of a PRA Upgrade versus PRA maintenance and update.	The Configuration Control Notebook specifies the difference between and update and an upgrade but does not specifically require performance of a peer review for upgrades. The Standard specifically calls for a peer review for PRA Upgrades, but the Configuration Control Notebook does not specifically call for one following an Upgrade. Modify the Configuration Control Notebook to specify that peer reviews are required for PRA Upgrades.	Section 3.3 of the configuration control notebook NB-PSA-CC [17] has been revised to include a requirement for a peer review against the ASME standard for PSA model upgrades. ILRT analysis – no impact.
MU-D1-01	Finding	The PRA configuration control process shall include evaluation of the impact of changes on previously implemented risk-informed decisions that have used the PRA AND that affect the safe operation of the plant.	The Configuration Control Notebook does not direct that updates or upgrades are compared with previous risk-informed decisions and have used the PRA. Review of previous RI applications is not called out in the Configuration Control Notebook. Add requirement for reviewing the previous RI applications against the new PRA results to see if they impact the results of the previous work.	Section 3.3 of the configuration control notebook NB-PSA-CC [17] has been revised to include requirements for the review of updates and upgrades against previous applications and analyses. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
QU-A3-01	Finding	ESTIMATE the mean CDF accounting for the state-of-knowledge correlation between event probabilities when significant [Note (1)].	The mean ISLOCA CDF frequency does not account for the state-of-knowledge correlation (SOKC). Per SR QU-A3, the effect of the SOKC has been found to be significant in cutsets contributing to ISLOCA frequency. Explicitly required in Note 1 of the SR. Update the ISLOCA frequencies with SOKC.	A method of demonstrating the effect of the state of knowledge is to perform a Monte Carlo simulation for representative cases. Given the reference to ISLOCA frequency in the ASME Standard and the finding, two examples were selected using as input the failure rate and distributions from the PSAR3 model, namely: ECCS injection line check valves FTRC and SDC MOVs FTRC. Each of these leads to an ISLOCA. Based on these simulations, a correction factor was applied as a recovery event to the ISLOCA cut sets generated by CAFTA containing the following components and failure modes when generating results with point factors: 2 MOVs FTRC - SOKC factor = 3 (SDC suction) 2 check valves FTRC - SOKC factor = 4 (HPSI and LPSI injection lines) 3 check valves FTRC - SOKC factor = 33 (HPSI injection lines) The result is an increase in the initiating event frequency from 2.04E-9 per year to 6.13E-9 per year. The LPSI injection and SDC lines dominate the ISLOCA results. A factor of 3 (SDC) or 4 (LPSI) is not a significant deviation, particularly for applications where uncertainty analyses are performed as a part of the evaluation. Because incorporating the suggested rules file into the SAPHIRE model results in a negligible impact on overall core damage frequency (within the uncertainty of the analysis), the event tree rules and basic events developed here to account for the SOKC will only be incorporated into the model for specific applications that examine ISLOCA events. ILRT analysis – no impact.
QU-B2-01	Finding	TRUNCATE accident sequences and associated system models at a sufficiently low cutoff value that dependencies associated with significant cutsets or accident sequences are not eliminated. NOTE: Truncation should be carefully assessed in cases where cutsets are merged to create a solution (e.g., where system level cutsets are merged to create sequence level cutsets).	Palisades used a truncation level of 1E-09 for quantification and conducted evaluation of convergence of the results down to a truncation level of 1E-12. The truncation should be set to 1E¬11 based on the Palisades definition of significant accident sequences.	Analyses have shown that a CDF truncation at 1.0E-10/year is judged to be appropriate for assessing the Palisades CDF for risk-informed applications. The change in CDF from 1.0E-10 to 1.0E-11 is less than 5%. This is consistent with the ASME PRA standard HLR-QU-B that states, "convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF or LERF, and the final change is less than 5% which indicate that a truncation of four orders of magnitude below the CDF is adequate for a high quality PRA". Note that the Palisades Level 2 containment phenomenological event tree analysis is typically evaluated at a 1E-15 truncation value. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
QU-C1-01	Finding	IDENTIFY cutsets with multiple HFEs that potentially impact significant accident sequences/ cutsets by requantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. The final quantification of these post-initiator HFEs may be done at the cutset level or saved sequence level.	Conditional HEPs were developed by Palisades for several HFEs and incorporated in the fault tree models. Some accident sequences revealed HFE combinations for which dependency between the HFEs has not been assessed and documented. While the Palisades model has been quantified and cut sets for accident sequences have been identified, the review and update of those sequences with respect to combinations of HFEs is not complete. Complete review and update of accident sequence cut sets relating to combinations of HFEs.	The complete detailed methodology for evaluating human error dependency was completed as described in HRA Notebook NB-PSA-HR Volume 1, Section 5.2 [25]. The general steps used in this analysis are as follows: 1. Run the base model with the post-initiator action failure event probabilities set to 1.0. 2. Identify the multiple human action combinations that appear in the cut sets. 3. Identify the risk significant combinations assuming complete dependence. 4. Perform a dependency analysis on the risk significant combinations and develop conditional probabilities for dependent actions. 5. Incorporate the dependent combinations in the fault trees of the PSA. To address the human action dependency issue with respect to CDF, Palisades developed a systematic approach that investigated a sufficient number of human actions to merit confidence that the impact of these dependencies have been thoroughly assessed and adequately represented in the PSA models. The approach is iterative and methodical. ILRT analysis – no impact.
QU-D1-01	Finding	REVIEW a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct.	The final model review has not been completed and documented. The final review of accident sequence results has not been completed and documented so that the reasonableness of the results can be verified. Palisades indicated that this review is required but not complete. This finding is being written against all of the QU-D supporting requirements as well as some QU-F requirements. Palisades needs to complete the formal review of accident sequence quantification results and make modifications as needed to address issues found in that review. The final results should then be documented in the corresponding notebooks.	The documentation of the final model results, validation etc. will comport to the guidelines cited and Entergy procedures referenced in PSA Notebook NB-PSA-CC [17]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SC-C3-01	Finding	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the development of success criteria.	Some Calculations associated with success criteria are not in the Palisades formal document control system. In addition, the basis for the LOCA size ranges are not included. The issue regarding the basis for the LOCA size definitions is briefly repeated in suggestion F&O SC-C2-01. The formal document control system is predicated on an approved licensing action (e.g., Submittal of a License Amendment Request for NFPA 805 or a power uprate). Therefore, some calculations are not formally added to the system until the final project action is complete. This leaves some calculations used to support PRA success criteria out of the system for some time and could result in lost or modified documentation that does not comport with the PRA results. Technical bases for the size ranges are not included in the success criteria definitions.	With regard to success criteria, the technical reference is documented in the event tree and success criteria notebook: PLP0247-07-0004.01R0, Palisades Nuclear Plant Thermal Hydraulic MAAP calculations (R-1551). Additional discussion and basis regarding LOCA size and frequency determination is contained in calculation EA-PSA-IE-00-0010, Revision 0,"Calculation of Initiating Event Frequencies in Accordance with CEOG Standards". EA-PSA-IE-00-0010 was included as a reference in notebook NB-PSA-ETSC to improve the discussion regarding the basis for the determination of LOCA size ranges. These references were included in the overall set of documents provided to the peer review team on 10/26/09. It is worth noting that during the Palisades PTS study [27], a separate effort was underway at NRC to review and revise the LOCA frequencies from NUREG/CR-5750 for use particularly in work associated with 10CFR50.46 but with applicability for other risk-informed applications such as the PTS project. There was a concern that the LOCA frequencies in NUREG/CR-5750 did not account for age-related factors important to deriving the frequencies and an expert elicitation effort at NRC was conducted to account for these adjustments. Examining just the piping contribution it was concluded by the NRC Expert Elicitation committee that the Palisades plant specific initiating event frequencies were nearly the same as that developed in the elicitation effort. Therefore no change was made to the Palisades values during the conduct of the PTS analysis. And the current Palisades small break LOCA frequency 2.26E-03/yr is approximately an order of magnitude greater than that reported in NUREG/CR-6928 mean value of 5.77E-04/yr. In summary the Palisades LOCA frequencies are well documented and validated. With respect to design processes, the site process for formal document control is being followed. There is not an elevated potential for lost or modified documentation that does not comport with the PRA results since the new PRA results are not formal results until the entire engineering change related to the submittal is complete. The PRA staff is required to follow the plants design authority rules. In conclusion, from a technical context and process assessment perspective this is not considered a finding. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-A13-01	Finding	INCLUDE those failures that can cause flow diversion pathways that result in failure to meet the system success criteria.	Currently a flow diversion pathway is modeled for the Containment Spray pumps failing due to a diversion through a failed other Containment Spray pump with a failed outboard check valve. Although this is a valid flow diversion pathway during the injection mode of operation, it is not a flow diversion pathway during recirculation since the "diverted" flow would be diverted to the suction of the HPI pump - which is where the outlet of a portion of the Containment Spray (CS) flow is supposed to go anyway. Because the HPI pumps flow rate is a function of the pressure in the containment, it does not matter which CS path provide the flow the pump, the total flow to/through HPI will not be impacted by the pathway. Therefore, the total flow from the operating CS pump to the CS spargers will also not be impacted. Current modeling results in unnecessary conservatism. Include this flow diversion pathway only for injection modes of operation and remove from the recirculation mode of operation.	Gates FLW-DIV-P54B&C-INJ, "FLOW DIVERSION TRHOUGH P-54B AND P-54C DURING INJECTION MODE," FLW-DIV-P54A&B-INJ "FLOW DIVERSION TRHOUGH P-54A AND P-54B DURING INJECTION MODE," and FLW-DIV-P54A&C-INJ "FLOW DIVERSION TRHOUGH P-54A AND P-54C DURING INJECTION MODE" were added to the PRA model (PSAR3 Fire [16]). These gates are coupled with house event ESS-HSE-RAS-PRE which is set to true for modeling fault trees applicable only to pre-RAS (injection) mode of operation. When true, the flow diversion results in no flow from the affected containment spray pump. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-A20-01	Finding	INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity (see DA-C14).	Palisades specifically models planned activities resulting in coincident unavailability of equipment in multiple trains of different systems that belong to similar divisions (such as train A of AFW and train A of HPI) but does not include events that might occur associated with coincident unavailability of multiple trains of different systems that belong to opposite division (such as train A of AFW and train B of HPI). Potential unavailability between systems involving opposite divisions due to planned activities is not included in the model and may result in non-conservative results. Include events in the model that address coincident unavailabilities associated with train A of one system with train B of another, redundant systems due to planned activities (if the experience shows any exist).	Coincident unavailability was re-evaluated and updated in Section 9.1 of the data analysis notebook, NB-PSA-DA [5]. From that evaluation: "To evaluate coincident unavailability, all the unavailability data was compiled, and coincident events were marked. Coincident unavailability was considered for each train (i.e., 2 or more train A components OOS at the same time), and for both trains (i.e., 1 or more Train A components OOS at the same time as 1 or more Train B components). In addition to reviewing the maintenance rule unavailability data for coincident unavailability, the risk management work week reviews from the LAN were also downloaded and reviewed. The following identifies the equipment associated with each train: • Train A equipment: C-2A & C-2C, C-6B, ED-15 & ED-17, K-6A, P-52C, P-54B & P-54C, P-55C, P-56A, P-66B, P-67B, P-7B, P-8A & P-8B, and PRV-1042. • Train B equipment: C-2B, C-6A, ED-16 & ED-18, K-6B, P-52B, P-54A, P-55A & P-55B, P-56B, P-66A, P-67A, P-7A & 7C, P-8C and PRV-1043. Plant experience showed that in most cases only one piece of equipment from a train is removed from service at a time. A review of the three plus years of unavailability data showed that there was limited, repetitive coincident unavailability; most cases involved only two components, and occurred only once in the three year data window. There were, however, a few cases in which plant experience showed that two components were recurrently removed from service at the same time. In these cases, coincident unavailability was modeled; the following identifies the combinations of equipment for coincident unavailability: 1. P-54B and P-66B; 2. P-54B and P-67B; 3. P-54C and P-67B; 4. P-8A and P-8B; 5. P-54A and P-66A; 6. P-54A and P-67A; and 7. P-56A and P-56B. Basic events were developed for items 1-7 above and documented in Attachment 12, Table 12.1 [5].

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-A20-01 (Cont'd)	Finding			Coincident unavailability included only the time that both components were simultaneously unavailable. If one component was unavailable for an extra hour, the hour was used in the individual unavailability. Once coincident unavailabilities were calculated, the times were subtracted from the individual unavailabilities to avoid double counting." ILRT analysis – no impact.
SY-B3-01	Finding	ESTABLISH common cause failure groups by using a logical, systematic process that considers similarity in (a) service conditions (b) environment (c) design or manufacturer (d) maintenance JUSTIFY the basis for selecting common cause component groups. Candidates for common cause failures include, for example: (a) motor-operated valves (b) pumps (c) safety-relief valves (d) air-operated valves (e) solenoid-operated valves (f) check valves (g) diesel generators (h) batteries (i) inverters and battery charger (j) circuit breakers	Common cause failures as a whole are modeled correctly and consistently. However, the modeling of the HPI, LPI, and common line check valves is producing non-minimal and potentially non-valid cutsets. Because of the safety significance of the LPI and HPI systems, the non-minimal and non-valid cutsets are overestimating the risk associated with those failures. Review the common cause modeling of components in the PRA model, especially of the valves in series and revise the model as appropriate. Alternatively, non-valid combinations can be added to the mutually exclusive file to remove the non-minimal and non-valid cutsets.	A full evaluation of this finding is presented in Attachment 1[9]. Examination of cut sets that include CCF of in-series components reveals that there are no non-minimal cut sets. Treating in-series HPSI and LPSI valves as independent (incorporating the CCF portion of the valve failure in the failure probability for each valve), as appears to be suggested by this finding, turns out to be the more conservative approach. The Palisades approach produces realistic and valid results. The modeling of common cause failures, as applied in the Palisades PRA, is based on, and consistent with, the Multiple Greek Letter approach. This approach produces valid cut sets, even if those cut sets may indicate that more components have failed than necessary. The approximation suggested by this finding is, in fact, a more conservative approach which can overestimate risk. If the beta factor is small, then this overestimation is not significant. The approximation used in the Palisades PRA, namely, using the "total" failure rate to represent the "independent" failure rate without correcting by the factor of (1-beta), also does not introduce significant conservatism in the results. Therefore, the concerns expressed by this finding do not appear to be correct, and modeling or quantification changes are not considered necessary. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B5-01	Finding	ACCOUNT explicitly for the modeled system's dependency on support systems or interfacing systems in the modeling process. This may be accomplished in one of the following ways: (a) for the fault tree linking approach by modeling the dependencies as a link to an appropriate event or gate in the support system fault tree; (b) for the linked event tree approach, by using event tree logic rules, or calculating a probability for each split fraction conditional on the scenario definition.	There is an apparent error in the EDG failure to run logic: this logic does not account for the SWS pump failures to start. When the PRA Group was shown the apparent error, they admitted that it was an error and that they had also identified it in their Self Assessment. The model was corrected while the Review Team was on-site, but a review of the affected cutsets still has a cutset with a diesel generator run failure in the same cutset as the SWS pump failure to start. Given failure of the SWS pump to start, the diesel generator fail to run should be 1.0. SWS pump failures to start are valid contributors to EDG failure. The model should account for these contributors and the diesel generator failures need to be adjusted to account for the availability of SWS These specific failures should be incorporated into the fault tree model. And, given the similarity of this finding with Finding SY-B5-02, it is recommended that a systematic review of other potentially risk important dependencies be performed.	This is not considered a finding. SWS start failures are captured under the diesel failure to start gates. Start, load/run and run failures are all captured under 'OR' gates so the logic is equivalent. The PRA model Release 2b cutsets properly account for diesel run and service water pump start failures. This issue was noted under supporting requirement QU-D5 in the Reg. Guide 1.200 Self Assessment (NB-PSA-SA Rev 0) for model Release 2a. It was subsequently corrected in Release 2b delivered on 10/26/09 and again noted in the updated Self Assessment [18]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B5-02	Finding	ACCOUNT explicitly for the modeled system's dependency on support systems or interfacing systems in the modeling process. This may be accomplished in one of the following ways: (a) for the fault tree linking approach by modeling the dependencies as a link to an appropriate event or gate in the support system fault tree; (b) for the linked event tree approach, by using event tree logic rules, or calculating a probability for each split fraction conditional on the scenario definition.	Potentially risk-significant manual valves were excluded from the model without explanation. Their exclusion should be based on SR SY-A15 screening criteria. For example, manual valves in the Containment Spray system flow paths were not modeled. It was noted that some of these manual valves are actually depicted on the simplified system drawings, but they are not labeled. To avoid confusion, it is suggested that all components in these drawings be labeled. Note: site practice is to include all mechanical components on the simplified PRA schematics and to label only those components specifically included. This provides a quick indication of what components are physically present but not explicitly modeled. Excluded manual valves may be risk significant. Provide explanation for the excluded valves based on SY-A15 or include them in the model.	Per supporting requirement SY-A15 [1]: A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. The valves described in finding SY-B5-02 in the containment spray system are normally locked open manual valves. The Palisades PRA has assumed that random failure or plugging of locked open manual valves are not a significant contribution to system or component failure, however, this assumption was not explicitly documented. Assumption number A-0047 was developed and added to the PRA assumptions database, success criteria notebook [11], and appropriate system notebooks. The assumptions states: "NUREG CR-6928 (January 2007) Table 5-1, provides data for manual valve failure to open and failure to close. Failure to remain open is not evaluated and no data is provided for this failure mode. Plugging has a mean failure probability of E-09. A valve locked in position, is very unlikely to be susceptible to environmental effects such as vibration that could result in valve closure. In addition, locked valves are strictly controlled by keys issued from the control room and systems with locked open valves are either normally in operation or are frequently tested to meet technical specification requirements. A mispositioned or plugged valve would likely be detected as part of plant operator rounds, testing, check lists, or data collection and would be promptly re-positioned or repaired as necessary. Based on the generic failure data for plugging, testing, and strict controls of these valves, the probability of valve failure is very small and would have a negligible impact on system failure rate. This assumption is not applicable to pre-initiator human error events where a valve is repositioned for testing or maintenance." With respect to manual valve misalignment, a scoping analysis was performed as documented in the Palisades HRA notebook NB-PSA-HR Volume 2, "Palisades Pre-initiator Human Error Evaluation" [8]. The results of this scoping demonstrated that a number of manual valves are susceptible to mispositioning with a non-significant failure probability. The basic events developed and scoping methodology are presented in that document.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
SY-B5-02 (Cont'd)	Finding			A sampling of locked open and non-locking manual valves were evaluated from the auxiliary feedwater, shutdown cooling, and atmospheric dump valve fault trees to provide validation that assumption A-0047 is applied consistently to all system fault trees. No discrepancies were found. ILRT analysis – no impact
SY-B11-01	Finding	MODEL the ability of the available inventories of air, power, and cooling to support the mission time.	The current model for the supplemental diesel model, however, is not completely correct as CB 152-203 should be "fails to remain closed" instead of "Fails to remain open," and failure of the A14 safeguards bus needs to be added to the model as a reason the Supplemental DG fails to provide power to the 1D Safeguards bus. The 152-203 CB is modeled under another portion of the logic for power to the 1D Safeguards bus. Using the incorrect failure mode/basic event for the CB failure results in the impact for the failure of the CB not being adequately captured in the model. Revise the modeling to correct the CB failure mode modeled, and add failure of the 1A bus itself to the model.	A new fault tree was created, PNOSGPWR "NO SAFEGUARDS POWER TO SAFEGUARDS BUS" that models failure of buses 1C, 1D, or 1E and failure of their respective breakers that tie them to the safeguards bus to open. This fault tree was placed under gates: • PNOSGPWR1D, NO SAFEGUARDS POWER TO BUS 1D • PNOSGPWR1C, NO POWER FROM SAFEGUARDS BUS TO BUS 1C • PNOSGPWR1E, NO SAFEGUARDS POWER TO BUS 1E Additional breaker failure logic was also added to model failures associated with the non-safety-related (NSR) diesel generator. This logic considers that the emergency diesel generator (EDG 1-2) and safety related bus supply breakers (1C and 1D) must open prior to starting the NSR diesel generator and subsequently re-closing to supply the appropriate bus. Under gate DG-NSR-START1D-03, "CIRCUIT BREAKER FAILURES" added basic events: • ACP-C2MA-152-213, CIRCUIT BREAKER 152-213 FAILS TO OPEN • ACP-C2MA-152-203, CIRCUIT BREAKER 152-203 FAILS TO OPEN • ACP-C2MB-152-203, CIRCUIT BREAKER 152-203 FAILS TO CLOSE

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B11-01 (Cont'd)	Finding			Under gate DG-NSR-RUN1D-03, "CIRCUIT BREAKER FAILURES" added basic events: • ACP-C2MC-152-203, CIRCUIT BREAKER 152-203 FAILS TO REMAIN CLOSED Under gate DG-NSR-START1C-03, "CIRCUIT BREAKER FAILURES" added basic event: • ACP-C2MB-152-403, NSR EDG OUTPUT BREAKER 152-403 FAILS TO CLOSE These logic changes capture all of the appropriate breaker failure modes related to the non-safety related emergency diesel generator and the safeguards bus. These model changes are documented in the model update to PSAR3 Fire [16]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B12-01	Finding	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system such as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.	Palisades did not model HVAC for the control room or the cable spreading room based on operator actions to implement alternate cooling strategies such as opening doors or using a proposed portable exhaust fans. (See pages 17 and 24 of attachment 8 to NB-PSA-ETSC r01). However, the operator actions to implement the alternate actions were not included in the models. There was never an intent to model the operator actions given that past analyses has shown that both rooms can survive a loss of HVAC. It is recognized that the analyses requires updating and that the documentation requires updating. Palisades should either provide additional justification for not modeling the HVAC systems for the cable spreading room and control room, or model the operator actions to implement alternate cooling strategies or model HVAC for these two rooms.	The basis for excluding control room HVAC from the full power internal events model was strengthened to include other aspects in addition to operator actions. This evaluation was fully documented in NB-PSA-ETSC [3]. The conclusion summary states: Control room cooling in the Palisades internal events PRA is not considered an issue based on the following: • the high design temperature limits of the major control room components, • the general conservative modeling assumptions employed throughout the EA-APR-95-023,R1 analysis, • the philosophy of the operators with respect to remaining in the control room during such an event, • the TMM's are not credited in the EOPs, • the relative un-importance of HVAC failure on a variety of plant PRA studies. Therefore it is considered unnecessary to model either loss of HVAC as an initiator or as a support system for the internal events model. An analysis of the cable spreading room heat-up following a loss of ventilation was developed using the GOTHIC software code [12]. The analysis approach was to establish the room's heat load based on Systematic Evaluation Program (SEP) Topic IX-5 (Phase II cable spreading room loss of HVAC testing) data by modeling the test boundary conditions in detail and iterating on room heat generation until the test results were mimicked by the model. With the room heat load established, the model boundary conditions were changed to establish a conservative scenario with no room ventilation. The room temperature profile demonstrated that at 48 hours the peak temperature would reach 122°F (50°C).

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B12-01 (Cont'd)	Finding			CALC-455-001-DC2 [13] was then performed to evaluate all cable spreading room equipment modeled in the PRA under these conditions. The analysis conservatively assumed that the room was at the peak calculated temperature of 122°F (50°C) for the entire 48 hour duration of the transient. An evaluation of equipment qualification reports, and vendor data, was then performed which concluded that reasonable assurance of operability is assured for all equipment at an elevated ambient temperature of 122°F for 48 hours. Based on the conclusions of these analyses, ventilation to the cable spreading area is not explicitly modeled as failure to re-establish ventilation does not result in equipment failure prior to the PRA 24 hour mission time. Attachment 8 of NB-PSA-ETSC [11] has been updated to reflect these conclusions. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
SY-B12-02	Finding	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system such as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.	Detailed analysis of systems/component Dependency on HVAC/ventilation should be provided in the individual systems and/or Dependency Tables. Palisades needs to provide better documentation of the basis for not modeling the HVAC within the system notebooks for the control room and cable spreading room. SWS pump failures to start are valid contributors to EDG failure. A review of the affected cutsets still has a cutset with a diesel generator run failure in the same cutset as the SWS pump failure to start. Given failure of the SWS pump to start, the diesel generator fail to run should be 1.0. The model needs to account for these and similar dependencies. These specific failures should be incorporated into the fault tree model. And, given the similarity of this finding with Finding SY-B5-02, it is recommended that a systematic review of other potentially risk important dependencies be performed.	The basis for excluding control room HVAC from the full power internal events model was strengthened to include other aspects in addition to operator actions. This evaluation was fully documented in NB-PSA-ETSC [11]. The conclusion summary states: Control room cooling in the Palisades internal events PRA is not considered an issue based on the following: • the high design temperature limits of the major control room components, • the general conservative modeling assumptions employed throughout the EA-APR-95-023,R1 analysis, • the philosophy of the operators with respect to remaining in the control room during such an event, • the TMM's are not credited in the EOPs, • the relative un-importance of HVAC failure on a variety of plant PRA studies. Therefore it is considered unnecessary to model either loss of HVAC as an initiator or as a support system for the internal events model. With respect to cable spreading room cooling. An analysis of the cable spreading room heat-up following a loss of ventilation was developed using the GOTHIC software code [12]. The analysis approach was to establish the room's heat load based on Systematic Evaluation Program (SEP) Topic IX-5 (Phase II cable spreading room loss of HVAC testing) data by modeling the test boundary conditions in detail and iterating on room heat generation until the test results were mimicked by the model. With the room heat load established, the model boundary conditions were changed to establish a conservative scenario with no room ventilation. The room temperature profile demonstrated that at 48 hours the peak temperature would reach 122°F (50°C).

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B12-02 (Cont'd)	Finding			CALC-455-001-DC2 [13] was then performed to evaluate all cable spreading room equipment modeled in the PRA under these conditions. The analysis conservatively assumed that the room was at the peak calculated temperature of 122°F (50°C) for the entire 48 hour duration of the transient. An evaluation of equipment qualification reports, and vendor data, was then performed which concluded that reasonable assurance of operability is assured for all equipment at an elevated ambient temperature of 122°F for 48 hours. Based on the conclusions of these analyses, ventilation to the cable spreading area is not explicitly modeled as failure to re-establish ventilation does not result in equipment failure prior to the PRA 24 hour mission time. Attachment 8 of NB-PSA-ETSC [11] has been updated to reflect these conclusions. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
AS-C2-02	Suggestion	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results. For example, this documentation typically includes: (a) the linkage between the modeled initiating event in the Initiating Event Analysis section and the accident sequence model; (b) the success criteria established for each modeled initiating event including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities); (c) a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems and operator actions, end states, and other pertinent information required to fully establish the sequence of events); (d) the operator actions reflected in the event trees, and the sequence-specific timing and dependencies that are traceable to the HRA for these actions; (e) the interface of the accident sequence models with plant damage states; (f) [when sequences are modeled using a single top event fault tree] the manner in which the requirements for accident sequence analysis have been satisfied.	While it is obvious that the Flag files exist, the development and review of the Flag file used in SAPHIRE is not included in the accident sequence analysis documentation. The Flag file does appear to be documented in the Quantification report. However, because this file governs how the accident sequences are quantified, it is important to ensure the accident sequences (especially the support system initiators) are handled correctly in the SAPHIRE model, that the model is modified correctly for applications, and is important for long term maintenance and update of the model. To support this, documentation of the Flag file is an important part of the accident sequence documentation. It is recommended that Palisades provide at least a brief discussion of the Flag and provide a link to the documentation as it exists in the quantification report.	A summary discussion of the event tree rules (Flag) file was added to Section 3.0 of the event trees and success criteria notebook NB-PSA-ETSC [11]. In addition, Table 3.0-1 was added which lists all of the initiating event logical variables set to true in the rules file for a given initiating transient event. The summary references the detailed discussion for developing event tree rules which is documented in Section 3.0 of the quantification notebook NB-PSA-QU [6]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
DA-C16-01	Suggestion	Data on recovery from loss of offsite power, loss of service water, etc. are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service.	The SR is met based on a review of data provided in Attachment 9 of NB-PSA-IE (Initiating Event Notebook). Table 9.1 of NB-PSA-IE lists Industry LOOP Events (1980-2008). The Attachment 9 text and Table 9.1 indicate that the August 14, 2003 North America LOOP events were included (even though it did not affect Palisades). However, given that this event was very long for many plants (and very long recoveries significantly affects the LOOP recovery distribution), additional discussion of its treatment is appropriate. In addition, given that other long-term LOOP events were screened as not applicable to Palisades, it is suggested that a sensitivity analysis address the effect of the screening process. Loss of off-site power is an important risk contributor and the effect of the screening of longer LOOP events in the LOOP recovery analysis have a significant impact on the risk. Document treatment of August 14, 2003 North America LOOP event and perform a sensitivity study.	Currently Palisades is re-evaluating the data analysis and the time dependent models for the treatment of LOOP events. The modeling aspects include the time of LOOP recovery, the time of onsite power system recovery, EDG mission time, and the coping time between the time of an SBO event and the time when electric power must be recovered to prevent core damage. In addition to analyzing these interactions, the August 2003 northeast blackout event is evaluated in the data analysis as well. Expected completion of this assessment is the first quarter of 2011. ILRT analysis – no impact.
HR-C1-01	Suggestion	For each unscreened activity, DEFINE a human failure event (HFE) that represents the impact of the human failure at the appropriate level, i.e., function, system, train, or component affected.)	Many of the pre-initiator human failure events identified in Tables E.2-1A and E.2-1B do not match the basic event name in the Palisades fault tree PSAR3 Release #2B.caf. For these events, it appears that the system designator has been expanded from one character to three characters in the BE name. Inconsistencies between the documentation and the model make reviews difficult and might lead to additional questions on model adequacy. Update the HRA evaluations and the HRA document to match the BEs listed in the fault tree.	The pre-initiator process was revised to include a process of assessing each system. The initial step of the HFE identification process was to identify the plant systems to be considered in the review. The Palisades pre-initiator methodology [8] indicates that the review should include all systems modeled in the PRA, which are listed in the Palisades System Notebooks. Once the initial systems list was assembled, the system descriptions and simplified P&IDs were examined to identify and define the Train/Function/Channel (TFC) for the system. Those TFCs not susceptible to Type A (pre-initiator) events were screened from further review (this process is documented in Table 2.2-1). For each of the unscreened TFCs identified, a scoping event was added to the PRA model. The scoping values were then used to determine the risk significance of each event and evaluate which events should remain in the model. The basis for any exclusion of pre-initiator for a system is documented in the updated HRA Notebook Volume 2. A specific check to confirm that basic events representing the pre-initiators was performed to confirm events in the PRA model agree with the development discussed in the HRA notebook [8]. ILRT analysis – no impact.

Table A.2.3-1				
F&O # (Supporting Requirement)	**Finding or Suggestion**	**ASME Reg. Guide 1.200 Category II Text**	**Finding Description (summary discussion)**	**Disposition**
IE-B4-01	Suggestion	GROUP separately from other initiating event categories those categories with different plant response (i.e., those with different success rate criteria) impacts or those that could have more severe radionuclide release potential (e.g., LERF). This includes such initiators as excessive LOCA, interfacing systems LOCA, steam generator tube ruptures, and unisolated breaks outside containment.	Palisades did not model excess LOCA such as random vessel rupture based on their Pressurized Thermal Shock (PTS) evaluation. However, an excessive LOCA event is explicitly called out in the SR. However, because of the low generic Initiating event frequency, this is not expected to have a significant impact on the results. Excessive LOCA/Vessel Rupture should be included in the model as leading directly to core damage. Palisades can use the generic frequency or they can use the frequency from their Pressurized Thermal Shock Analysis can be used.	Palisades was one of three pilot plants evaluated in the recent NRC effort to re-evaluate the risk of pressurized thermal shock. These efforts are summarized in NUREG-1806 and NUREG-1874. The analyses made use of three Palisades specific analytical models (PRA, RELAP and FAVOR) that together, allowed the estimate of the yearly through-wall crack frequency (TWCF) in a reactor pressure vessel (RPV). Using the 20+ year old NUREG-1150 data ("the generic frequency") to model Excessive LOCA/ Vessel Rupture in lieu of the latest plant specific state-of-knowledge based on the joint RES/Industry PTS initiative is not warranted. Note that the dominate sequence was a non-mechanistic scenario that assumed the pressurizer safeties failed open for a period of time and subsequently reclosed. The next set of dominant sequences did not include a pressure component. Refer to NB-PSA-IE [10]. NB-PSA-IE dedicates 4 pages addressing Pressurized Thermal Shock. Palisades was one of three pilot plants evaluated in the NRC initiative to re-evaluate the risk of pressurized thermal shock. The analyses made use of three Palisades specific analytical models (PRA, RELAP and FAVOR) that together, allowed the estimate of the yearly through-wall crack frequency (TWCF) in the reactor pressure vessel (RPV). Using the 20+ year old NUREG-1150 data ("the generic frequency") to model Excessive LOCA/ Vessel Rupture in lieu of the latest plant specific state-of-knowledge based on the joint RES/Industry 50.61 initiative is not warranted. Note the dominant sequence was a non-mechanistic scenario that assumed the pressurizer safeties failed open for a period of time and subsequently re-closed. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
IE-C1-01	Suggestion	CALCULATE the initiating event frequency accounting for relevant generic and plant-specific data unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. (See also IE-C13 for requirements for rare and extremely rare events).	This is in Reference to 3.1.2: "The thermal capacity of the steam generators at Palisades is such that a demand on the PORVs or pressurizer SRVs is not expected following a reactor trip. This has been validated per review of past thermal hydraulic analyses (Final Safety Analysis Report (FSAR) Chapter 14). In addition, in the 30 plus years of operation, the plant has not experienced such an event. Moreover, the Palisades nominal operating pressure of 2060 psia is about 100 psi less than that of all PWRs. Only inadvertent or premature operation of these valves can lead to loss of coolant type conditions. Given a demand and subsequent failure of the pressurizer SRV's, the consequences of a small break LOCA are analyzed by linking to a replication of the baseline small break LOCA event tree. Although Palisades has not experienced pressurizer safety valve setup or setpoint drift problems, operating experience [e.g., Fort Calhoun (Licensee Event Report (LER) 285/92-028) and Calvert Cliffs (LER 317/94-007)] has shown that such events are plausible. As such, this event has been included in the model (EA-PSA-PSAR2-04-02)." Ensure that the Palisades definition for IE-LOCA-PZRSRV is consistent with the definition and events used to calculate NUREG/CR-6928's for IE-SORV (PWR). The events in NUREG/CR¬6928 IE-SORV were used directly in defining the prior. but are actually consequential SORV following another initiating event versus a spurious opening of a relief valve. Documentation could be improved.	Section 3.1 of the initiating events notebook NB-PSA-IE [10] states that this event models a transient demand on a pressurizer safety relief valve. Section 5.9 of NB-PSA-IE was revised to explicitly state that this event is treated as a conditional probability following an initiating event. The frequency for this event was calculated by dividing the IE frequency by the sum of all other initiating events. The definition of IE-SORV (PWR) from NUREG/CR-6928 is presented in this section to clarify the basis for the frequency development of this event and how it was applied in the model. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
IE-C12-01	Suggestion	COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonableness check of the results.	In Section 5 of NB-PSA-IEr1 describes the quantification of the Initiating Event Frequencies. As part of this quantification, data from numerous outside sources and is combined with plant specific data via Bayesian updates. However, there was no comparison of the results to the frequencies used by other similar plants. The SR requires a reasonableness check of the initiating event frequencies against those of other plants. While Palisades did do comparisons against generic data, there was no plant to plant comparison in most cases. Palisades should include a table showing their initiating event frequencies and the equivalent frequencies for one or more plants of similar vintage. Where there are large differences, Palisades should explain and justify the differences.	Table 5.15 was added to Section 5.12 of the initiating events notebook NB-PSA-IE [10]. This table presents a comparison of Palisades initiating events and frequencies to those developed at Waterford 3, which is a similar Combustion Engineering designed PWR. Where significant differences are noted the table provides additional notes. In addition, LOCA IE frequency validation occurred during conduct of the Palisades pressurized thermal shock (PTS) analyses. There was a concern that the LOCA frequencies in NUREG/CR-5750 did not account for age-related factors important to deriving the frequencies. An expert elicitation effort (independent of RES) at the NRC was conducted to account for these adjustments. The NRC expert elicitation subject matter experts concluded that the Palisades plant specific initiating event frequencies (employed in the 50.61 RES / Industry initiative and used in the current internal events analysis) were nearly the same as that developed in the elicitation effort. Therefore no change was made to the Palisades LOCA IE values. ILRT analysis – no impact.
MU-B3-01	Suggestion	PRA changes shall be performed consistent with the previously defined Supporting Requirements.	Section 6.2 of the Configuration Control Notebook requires review of model revisions to ensure that they appropriately implemented. The configuration control document does not specifically indicate that updates are to be done in accordance with corresponding SRs from the standard, but it is assumed that the definition of "appropriately implemented" includes such as review because the associated system, IE, or other notebooks that would be updated all currently have a section for self assessment against the standard. Add a sentence to the configuration control document to clarify that "appropriately implemented" means conformance to the standard supporting requirements.	Sections 3.3 and 6.2 of the configuration control notebook NB-PSA-CC [17] have been revised to include a requirement for the review of updates and upgrades against the ASME standard. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SC-A5-01	Suggestion	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For example, if following a LOCA, low pressure injection is available for 1 hour, after which recirculation is required, the mission time for LPSI may be 1 hour and the mission time for recirculation may be 23 hours. For sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, PERFORM additional evaluation or modeling by using an appropriate technique. Examples of appropriate techniques include: (a) assigning an appropriate plant damage state for the sequence; (b) extending the mission time, and adjusting the affected analyses, to the point at which conditions can be shown to reach acceptable values; or (c) modeling additional system recovery or operator actions for the sequence, in accordance with requirements stated in Systems Analysis (2-2.4) and Human Reliability (2-2.5) to demonstrate that a successful outcome is achieved.	Palisades uses 24 hours as the default mission time for all sequences that end in a stable end state. This can be potentially overly conservative for some sequences such as LOOP sequences when power is not recovered by 4 hours. A recovery factor considering the convolution of EDG FTR with offsite power was used but did not account for increased time for recovery as a function of the time that the EDG could run before failure. Using 24 hours for FTR in some sequences overestimates the importance of some events. Potentially adjust the EDG FTR recovery factor to credit the increased time available for recovery of offsite power as a function of how long the EDG runs before failure.	Currently Palisades is re-evaluating the data analysis and the time dependent models for the treatment of LOOP events. The modeling aspects include the time of LOOP recovery, the time of onsite power system recovery, EDG mission time, and the coping time between the time of an SBO event and the time when electric power must be recovered to prevent core damage. In addition to analyzing these interactions, the August 2003 northeast blackout event is evaluated in the data analysis as well. Expected completion of this assessment is the first quarter of 2011. ILRT analysis – no impact. Note that preliminary results indicate the numbers will be more favorable.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SC-B5-01	Suggestion	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include: (a) comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features (b) comparison with results of similar analyses performed with other plant-specific codes (c) check by other means appropriate to the particular analysis	Although the success criteria appear to be reasonable and consistent, there was no documented evidence that they had been checked against generic or other plants. Palisades did provide some documentation on how the success criteria were developed and how they compared to Combustion Engineering Owners Group guidance but there was no single, centralized set of documentation to demonstrate how Palisades met the comparison requirement of the SR. Palisades needs to provide documentation of the comparison to other generic or similar plants or provide a set of references to other documents that support this requirement.	Section 10.0 and Table 10.0-1 were inserted in notebook NB-PSA-ETSC [11]. This section describes a comparison of the Palisades success criteria to some comparable event tree headings developed for Waterford 3, which is a similar Combustion Engineering designed PWR. The review concludes that there are no significant outliers in the success criteria between the two plants that cannot be attributed to design differences. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SC-C2-01	Suggestion	DOCUMENT the processes used to develop overall PRA success criteria and the supporting engineering bases, including the inputs, methods, and results. For example, this documentation typically includes: (a) the definition of core damage used in the PRA including the bases for any selected parameter value used in the definition (e.g., peak cladding temperature or reactor vessel level) (b) calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used (c) identification of computer codes or other methods used to establish plant-specific success criteria (d) a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) of the calculations or codes (e) the uses of expert judgment within the PRA, and rationale for such uses (f) a summary of success criteria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA (g) the basis for establishing the time available for human actions (h) descriptions of processes used to define success criteria for grouped initiating events or accident sequences	LOCA break sizes are given in detail. However, the traceability of the references provided for where and how these break sizes were determined is difficult to follow to the ultimate basis.. Based on discussion with Brian Brogan, a reference is available for these break sizes. Documentation only. Include a reference in the success criteria notebook that shows how the LOCA break sizes were determined.	The primary technical basis reference is included in the event tree and success criteria notebook: PLP0247-07-0004.01R0, "Palisades Nuclear Plant Thermal Hydraulic MAAP Calculations" [14]. Additional description and technical basis is contained in calculation EA-PSA-IE-00-0010 [19], "Calculation of Initiating Event Frequencies in Accordance with CEOG Standards". These references were added to Section 5.0 of NB-PSA-ETSC [11]. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B14-01	Suggestion	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include (a) LOCA inside containment with failure of containment heat removal (b) safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) (c) steam line breaks outside containment (d) debris that could plug screens/filters (both internal and external to the plant) (e) heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability (f) loss of NPSH for pumps (g) steam binding of pumps.	One potential weakness identified is the documentation and handling of the Containment Sump Blockage potential. A discussion of the sump blockage potential was not found in the SSS notebook, and a common cause sump blockage event was not found in the associated fault tree model. Note: independent sump blockage events are included in the model. Because of the significance and impact of the sump blockage potential, the impact of this issue should be discussed in the system notebook and included in the model as appropriate. (Note: Palisades did identify this issue in their self-assessment but it remained unresolved at the time of the peer review.) Include a discussion of the sump blockage potential issue in the SSS notebook, including the discussion of the inclusion/or exclusion for common cause blockage of the strainers, and revise the fault tree model as appropriate.	Common cause sump blockage events were added to the model (see gates CCF-316 and CCF-317) and documentation updated (NB-PSA-SY-SSS Section 2.5 [21] and NB-PSA-SM, Table 5.10-1 [20]. Description of sump strainer and discussion of sump blockage was added to the SIRWT Tank and Containment Sump Suction System notebook (NB-PSA-SY-SSS, Sections 1.0, 1.1 and 2.12). ILRT analysis – no impact.
SY-B15-01	Suggestion	INCLUDE operator interface dependencies across systems or trains, where applicable.	In NB-PSA-CSS, On p24, there is a statement that two human actions are modeled, CSS-Door-167 and CCS-Door-167B and pointed to Attachment B. Attachment B in turn pointed to a file Entitled CCC System Human Failure Event Table. This table contained only one event, CCS-PCMT-POC-0909. Discussions with Palisades PRA personnel indicate that the references on page 24 were old references pertaining to a sensitivity cases on the impact of leaving the CSS doors open during a steam line break. Typo only. Palisades needs to clean up these references.	The door events are not modeled as a probability per year that the specific door is in the open state, and are not considered human failure events (EA-PSA-CCW-HELB-02-17 [22]). Updated Sections 2.6 and 2.7 of the component cooling system notebook, NB-PSA-SY-CCS [28], to point to the correct reference. ILRT analysis – no impact.

Table A.2.3-1

F&O # (Supporting Requirement)	Finding or Suggestion	ASME Reg. Guide 1.200 Category II Text	Finding Description (summary discussion)	Disposition
SY-B4-01	Suggestion	INCORPORATE common cause failures into the system model consistent with the common cause model used for data analysis. (See DA-D6.)	Because the CCF modeling approach for CCCGs greater than 5 is bounding, it is recommended that the impact of this conservatism be investigated in the sensitivity analysis.	An evaluation of this peer review team suggestion was performed in Attachment 4 [9]. In summary, the evaluation concludes the following: • The global CCF factor chosen for this evaluation for 8 components failing due to common cause has a value that is higher (by a factor of more than 50) than the factor that is calculated using the explicit multiple greek letter (MGL) approach. • At the single component level, both the global and explicit approaches produce the same result. In other words, the bounding value of the global CCF factor is representative of the excluded combinations of components, and vice versa. This is the expected result. • At the system level, the quantification of a "system" fault tree that incorporates the bounding global CCF event in addition to all other random and common cause failures that contribute to system failure suggests that the use of the global CCF factor is bounding, but small (on the order of several percent). • For a global CCF to have a significant effect on the overall results of the PRA, it likely would need to have an effect on multiple redundant systems. Such global CCF events may exist in the Palisades PRA (e.g., station power transformers, sequencers) and should be examined for potential further refinement of the CCF when they impact the results of an application significantly. The impact of the bounding approach is to produce results that are conservative, with the amount of conservatism being a function of the component group and its MGL factors. ILRT analysis – no impact.

A.3 Identification of Key Assumptions

The methodology employed in this risk assessment followed the NEI guidance. The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analysis indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.4 Summary

A PRA technical adequacy evaluation was performed consistent with the requirements of RG-1.200, Revision 1. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed extension to the ILRT interval for Palisades to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.5 References

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

[2] Combustion Engineering Owners Group (CEOG) , "Industry Peer Review the Probabilistic Safety Analysis (PSA) against the Combustion Engineering Owners Group (CEOG) PSA checklists', RIE 2000-02, CE-NPSD-1194-P Task 1037.

[3] ERIN Engineering and Research Inc., "PALISADES GAP ANALYSIS REVIEW AND UPDATE," P0495060007-2711-061215, October 2004.

[4] From David Finnicum to Bradford Grimmel, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements For The Palisades Nuclear Power Plant Probabilistic Risk Assessment," LTR-RAM-II-10-015, March 12, 2010.

[5] Palisades PSA Notebook NB-PSA-DA Rev. 5, "Palisades PSA Data Notebook".

[6] Palisades PSA Notebook NB-PSA-QU Rev. 2, "Quantification Guideline".

[7] EA-PSA-1999-010 Rev 0, "Palisades PSA Bayesian Update".

[8] Palisades PSA Notebook NB-PSA-HR Rev. 3, Palisades Human Reliability Analysis Notebook Volume 2 (Pre Initiator Operator Actions).

[9] EA-PSA-RG1.200F&O-10-01 Rev. 0, "Resolution of Reg. Guide 1.200 October 2009 Full Power Internal Events Peer Review Findings and Observations".

[10] Palisades PSA Notebook NB-PSA-IE Rev. 3, "Initiating Event Notebook".

[11] Palisades PSA Notebook NB-PSA-ETSC Rev. 2, "Event Trees and Success Criteria".

[12] EA-PSA-GOTHIC-CSRHEATUP-09-09 Rev. 0, "GOTHIC Cable Spreading Room Heat-Up".

[13] CALC-455-001-DC2 Rev. 0, "Evaluation of Equipment in the CSR when Exposed to Elevated Temperatures for 48 Hours".

[14] PLP0247-07-0004.01 Rev. 1, "Palisades Nuclear Plant Thermal Hydraulic MAAP Calculations".

[15] CALC-455-001-DC1 Rev. 0, "Survivability of Equipment inside Containment Following a PRA LOCA/MSLB".

[16] EA-PSA-PSAR3-FIRE-10-02 Rev. 0, "NFPA 805 PRA Modeling, Update of PSAR2c to PSAR3 Fire".

[17] Palisades PSA Notebook NB-PSA-CC Rev. 1, "PSA Model Configuration Control".

[18] Palisades PSA Notebook NB-PSA-SA Rev. 1, "RG 1.200 PRA Self-Assessment Against the ASME PRA Standard Requirements".

[19] EA-PSA-IE-00-0010 Rev. 0, "Calculation of Initiating Event Frequencies in Accordance with CEOG Standards".

[20] Palisades PSA Notebook NB-PSA-SM Rev. 2, "PSA Model Summary".

[21] Palisades PSA System Notebook NB-PSA-SY-SSS Rev. 1, "SIRW Tank and Containment Sump Suction System".

[22] EA-PSA-CCW-HELB-02-17 Rev. 0, "Evaluation of the Impact of a High Energy Line Break in CCW Room with either Door 167 to 590 Corridor Auxiliary Building or 167B to the West Engineered Safeguards Room Open".

[23] Palisades PSA Notebook NB-PSA-SS Rev. 0, "Palisades Safe and Stable States".

[24] Nuclear Management Company, "Update of Palisades CDF Model – PSAR2b to PSAR2c," Calculation No. EA-PSA-PSAR2c-06-10, Rev. 0, June 2006.

[25] Palisades PSA Notebook NB-PSA-HR Rev. 3, Palisades Human Reliability Analysis Notebook Volume 1 (Post Initiator Operator Actions).

[26] Palisades PSA Notebook NB-PSA-LE Rev. 1, "Level 2 Notebook".

[27] "Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," ADAMS Accession number ML042880473, October 6, 2004.

[28] Palisades PSA Notebook NB-PSA-SY-CCS Rev. 1, "Component Cooling System".

Attachment 5

List of Regulatory Commitments

One page follows

Attachment 5

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
PLP will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate		X	Upon NRC approval of this License Amendment Request (LAR)