HOLTEC PALISADES

27780 Blue Star Highway, Covert, MI 49043

PNP 2025-002

10 CFR 50.90

February 5, 2025

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

Palisades Nuclear Plant
NRC Docket No. 50-255
Renewed Facility Operating License No. DPR-20

Subject: License Amendment Request to Include Leak Before Break Methodology for Primary Coolant System Hot and Cold Leg Piping in Palisades Licensing Basis

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, *Application for amendment of license, construction permit, or early site permit*, Holtec[1] , on behalf of Holtec Palisades, LLC (Holtec Palisades), hereby requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) licensing basis to include Leak Before Break (LBB) methodology for Primary Coolant System (PCS) hot and cold leg loop piping.

The basis for this request is the approved Combustion Engineering Owner's Group (CEOG) evaluation CEN-367-A, *Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems* (Reference 5), which demonstrates that if a crack were to occur in PCS loop piping it would be detectable, remain stable, and not result in a guillotine or unstable slot break.

The NRC Safety Evaluation Report (SER) associated with CEN-367-A states: "when referencing this CEOG topical report as a technical basis for applying LBB to primary loop piping, licensees must submit information to demonstrate that leakage detection systems installed at the specific facility are consistent with Regulatory Guide 1.45." The PNP PCS leakage detection systems were previously evaluated by the NRC against RG 1.45 Revision 0 (References 6) as documented in References 7 and 8. Since that evaluation, an updated Revision 1 of RG 1.45 was promulgated (Reference 9) and several enhancements have been made to the PNP leakage detection systems including implementation of a PCS leak rate monitoring program based on industry best practices (Reference 10). As such, this LAR includes a comparison of the PNP leak rate monitoring program and installed leak detection systems against the regulatory positions described in Regulatory Guide 1.45, Revision 1.

By letter dated June 13, 2022 (Reference 2), Entergy Nuclear Operations, Inc. notified the NRC under 10 CFR 50.82, *Termination of license*, that it had permanently ceased operations and permanently removed fuel from the reactor vessel at PNP. Upon docketing the 10CFR50.82(a)(1) certifications, 10 CFR 50.82(a)(2) no longer authorizes operation of the

[1] Holtec Palisades, LLC ("Holtec Palisades") is the licensed owner of PNP. Holtec Decommissioning International, LLC ("HDI") is the licensed operator of PNP while the facility is in decommissioning. Pursuant to the license transfer application submitted in connection with the PNP restart (Reference 11), licensed authority will transfer from HDI to Palisades Energy, LLC ("Palisades Energy") upon NRC's approval of the transition from decommissioning back to power operations. Holtec Palisades will remain the licensed owner of PNP.

reactor, or emplacement or retention of fuel into the reactor vessel. However, shortly after PNP transitioned to a decommissioning facility, Holtec Palisades assumed ownership of PNP (Reference 3) and given the support from the Governor of the State of Michigan, Holtec commenced a project to return PNP to a power operations plant. The regulatory path to reauthorize power operations at PNP is described in Holtec letter dated March 13, 2023 (Reference 4). In order to operate the PNP reactor, Holtec also submitted a LAR to revise PNP Renewed Facility Operating License (RFOL) DPR-20. The LAR would revise RFOL, Appendix A, *Permanently Defueled Technical Specifications (PDTS),* to reflect the resumption of power operations at PNP (Reference 1). This proposed LAR does not revise the Palisades PDTS or proposed Technical Specifications (TS). Upon approval of the LAR, Palisades TS Bases and Updated Final Safety Analysis Report (UFSAR) will be revised as discussed in the LAR.

Holtec is currently targeting the implementation of this LAR in the third quarter of 2025. To support this schedule, Holtec respectfully requests that the NRC review the enclosed LAR on a schedule that will permit approval of the proposed LAR by August 15, 2025, and that the proposed amendment become effective upon docketing the notification of transition to power operations letter per Reference 1, with a 30-day implementation period.

The enclosure with this letter provides a detailed description and evaluation of the proposed changes to PNP licensing basis. The attachment to the enclosure contains tables summarizing PNP leakage detection system components.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a), *Notice for public comment*, subparagraph (1), using the standards in 10 CFR 50.92, *Issuance of amendment*, paragraph (c), and it has been determined that the changes involve no significant hazards consideration. The basis for this determination is included in Enclosure 1.

In accordance with 10 CFR 50.91(b), *State consultation*, Holtec is notifying the State of Michigan of this proposed license amendment by transmitting a copy of this letter, with its enclosures, to the designated State of Michigan official.

If you have any questions regarding this submittal, please contact Amy Filbrandt, Acting Regulatory Assurance Manager, at (269) 764-2520.

This letter contains no new regulatory commitments and no revisions to existing regulatory commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 5, 2025.

Respectfully,

Jean A. Fleming

Digitally signed by Jean A. Fleming
DN: cn=Jean A. Fleming, o=Holtec Decommissioning International, LLC, ou=Regulatory and Environmental Affairs, email=J.Fleming@Holtec.com
Date: 2025.02.05 14:34:55 -05'00'

Jean A. Fleming
Vice President, of Licensing, Regulatory Affairs & PSA
Holtec International

References:

1. Holtec Decommissioning International, LLC (HDI) letter to U.S. Nuclear Regulatory Commission (NRC), "License Amendment Request to Revise Renewed Facility Operating License and Permanently Defueled Technical Specifications to Support Resumption of Power Operations," dated December 14, 2023 (ADAMS Accession No. ML23348A148)

2. Entergy Nuclear Operations, Inc. letter to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated June 13, 2022 (ADAMS Accession No. ML22164A067)

3. NRC letter to HDI, "Palisades Nuclear Plant and Big Rock Point Plant – Issuance of Amendment Nos. 129 and 273 RE: Order Approving Transfer of Licenses and Conforming Administrative License Amendments," dated June 28, 2022 (ADAMS Accession No. ML22173A173)

4. HDI letter to NRC, "Regulatory Path to Reauthorize Power Operations at the Palisades Nuclear Plant," dated March 13, 2023 (ADAMS Accession No. ML23072A404)

5. CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems," dated February 1991 (ADAMS Accession No. ML20070S390)

6. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 0, May 1973 (ADAMS Accession No. ML003740113)

7. NRC letter to Consumers Power Company, "Palisades – SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection," dated February 4, 1982 (ADAMS Accession No. ML18046B254)

8. NUREG-0820, "Integrated Plant Safety Assessment Systematic Evaluation Program Palisades Plant Final Report," dated October 1982 (ADAMS Accession No. ML18047A670)

9. Regulatory Guide 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Revision 1, dated May 2008 (ADAMS Accession No. ML073200271)

10. Westinghouse PWR Owners Group Letter OG-07-286, "Recommendations for Implementation of Guidelines Defined in 'Standard Process and Methods for Calculating RCS Leak Rate for Pressurized Water Reactors' (WCAP-16423-NP, Rev 0) and 'Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors' (WCAP-16465-NP, Rev 0) (PA-OSC-0189 and PA-OSC-0218) with respect to NEI-03-08," dated October 19, 2006 (ADAMS Accession No. ML070310081)

11. Holtec Decommissioning International, LLC letter to U.S. Nuclear Regulatory Commission, “Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments,” dated December 6, 2023 (ADAMS Accession No. ML23340A161)

Enclosure: Description and Evaluation of Proposed Changes
Enclosure Attachment: RCPB Leakage Detection Systems RG 1.45 Requirements

cc: NRC Region III Regional Administrator
NRC Senior Resident Inspector – Palisades Nuclear Plant
NRC Project Manager – Palisades Nuclear Plant
Designated Michigan State Official

Enclosure

PNP 2025-002

Description and Evaluation of Proposed Changes

Enclosure to Palisades Letter PNP 2025-002
Evaluation of Proposed Changes

Subject: Request for Approval of Leak Before Break (LBB) Methodology for the Primary Coolant System (PCS) Piping.

1 SUMMARY DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), 50.90, *Application for amendment of license, construction permit, or early site permit*, Holtec[1], on behalf of Holtec Palisades, LLC (Holtec Palisades), hereby requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) licensing basis to credit Leak Before Break (LBB). This LAR is consistent with the LAR, dated December 14, 2023 (Reference 1), to revise the Permanently Defueled Technical Specifications (PDTS) to reflect the resumption of power operations at PNP, which is currently under NRC review. This LAR proposes no Technical Specification (TS) changes; however, Licensing Bases changes will be made under the 10 CFR 50.59, *Changes, tests and experiments,* review process upon LAR approval.

Holtec is currently targeting the implementation of this LAR in the third quarter of 2025. To support this schedule, Holtec respectfully requests that the NRC review the enclosed LAR on a schedule that will permit approval of the proposed LAR by August 15, 2025, and that the proposed amendment become effective upon docketing the notification of transition to power operations letter per Reference 1, with a 30-day implementation period.

The TS references in this LAR are based on the Power Operation Technical Specifications (POTS) submitted with Reference 1. The Updated Final Safety Analysis Report (UFSAR) references in this LAR are based on UFSAR revision 35 (Reference 3). Changes made to the UFSAR after Revision 35 related to this amendment will be evaluated as required by 10 CFR 50.59. Plant procedures referenced by this LAR are to those versions applicable to power operations after PNP transitions to a Power Operations Licensing Basis (POLB) per Reference 1.

Note that Combustion Engineering Owners Group (CEOG) and NRC references typically use the phrases Reactor Coolant System (RCS) and Reactor Coolant Pressure Boundary (RCPB), whereas Palisades uses the equivalent phrases Primary Coolant System (PCS) and Primary Coolant Pressure Boundary (PCPB), respectively. In this LAR, RCS is used interchangeably with PCS and RCPB is used interchangeably with PCPB, depending on which organization originated the document being referenced.

This LAR requests approval for application of a leak-before-break (LBB) methodology to piping for the large bore PCS piping at PNP. This proposed change would eliminate the need to account for the dynamic effects associated with high-energy pipe rupture in the PCS from the licensing and design bases of the PNP consistent with 10 CFR 50, Appendix A, General Design Criteria (GDC) 4.

[1] Holtec Palisades, LLC ("Holtec Palisades") is the licensed owner of PNP. Holtec Decommissioning International, LLC ("HDI") is the licensed operator of PNP while the facility is in decommissioning. Pursuant to the license transfer application submitted in connection with the PNP restart (Reference 25), licensed authority will transfer from HDI to Palisades Energy, LLC ("Palisades Energy") upon NRC's approval of the transition from decommissioning back to power operations. Holtec Palisades will remain the licensed owner of PNP.

With the approval to eliminate PCS pipe break loads, the requirement to design against a double guillotine break will translate to additional margin in piping analysis. Holtec desires the additional margin in the piping, pipe support, and component support analyses to eliminate future needs for additional pipe restraints or supports whose installation and maintenance could require personnel radiation exposure as well as resource expenditures to engineer, install, and maintain them. Holtec intends to apply LBB to the Reactor Vessel Internals (RVI) applied loads (specifically to the core shroud bolts) as an initial application.

The basis for this request is the approved Combustion Engineering Owner's Group (CEOG) evaluation CEN-367-A, *Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems* (Reference 2), which demonstrates that if a crack were to occur in PCS loop piping it would be detectable, remain stable, and not result in a guillotine or unstable slot break.

The submitted version of the CEOG evaluation (CEN-367) was provided as the technical basis for approval of a limited application of LBB at Palisades in Reference 20 for the High Thermal Performance (HTP^{TM}) fuel spacer grid loads before it was officially approved in Reference 2. Other precedents for LBB approval for other CEOG plants are included in this document in Section 4.2.

The NRC Safety Evaluation Report (SER) associated with CEN-367-A states: "when referencing this CEOG topical report as a technical basis for applying LBB to primary loop piping, licensees must submit information to demonstrate that leakage detection systems installed at the specific facility are consistent with Regulatory Guide 1.45." The Palisades PCS leakage detection system was previously evaluated by the NRC against Regulatory Guide (RG) 1.45, *Guidance on Monitoring and Responding to Reactor Coolant System Leakage*, Revision 0 (Reference 5) as documented in References 10 and 12. Since that evaluation, an updated revision 1 of RG 1.45 was promulgated (Reference 18) and several enhancements have been made to the Palisades leakage detection systems, including implementation of a PCS leak rate monitoring program based on industry best practices (Reference 19). As such, this LAR includes a comparison of the Palisades leak rate monitoring program and installed leak detection system against the regulatory positions described in Reference 18.

The NRC letter approving CEN-367-A includes the statement, "We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved." Thus, the evaluation herein focusses on applicability to Palisades. It was determined in the Section 3 evaluation of the LBB application to Palisades that this evaluation is compliant with NUREG-0800, Section 3.6.3, *Leak-Before-Break Evaluation Procedures*. This is determined based on the following criteria:

1. That water hammer, corrosion, creep, fatigue, erosion, environmental conditions, and indirect sources are remote causes of pipe rupture,
2. That a deterministic fracture mechanics evaluation has been completed and approved by the staff, and
3. That leak detection systems are sufficiently reliable, redundant, diverse and sensitive, and that margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

The LAR addresses the applicable regulatory evaluation in Section 4, including applicable regulatory requirements, industry precedents, and the No Significant Hazards consideration.

Environmental considerations are discussed in Section 5. It was determined that no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

2 DETAILED DESCRIPTION

2.1 System Design and Operation

The basis for this request is the approved Combustion Engineering Owner's Group (CEOG) evaluation CEN-367-A, *Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems* (Reference 2), which demonstrates that if a crack were to occur in PCS loop piping it would be detectable, remain stable, and not result in a guillotine or unstable slot break.

The PNP UFSAR (Reference 3) Section 4.3.6 describes the PCS piping. The primary coolant piping consists of lengths of 42-inch inside diameter (ID) hot leg pipe from the reactor vessel outlet to the steam generator inlet and lengths of 30-inch ID cold leg pipe between the steam generator outlet and the primary coolant pump (PCP) suction nozzle and between the PCP discharge and the reactor vessel inlets. The primary coolant piping is of rolled bond clad plate construction, having a base metal of ASTM A 516, Grade 70, with a cladding of 304L stainless steel with a nominal thickness of 1/4 inch.

The UFSAR (Reference 3), Table 4-1, lists the PCS design and operating parameters at PNP. The design PCS temperature and pressure is 650°F and 2500 psia, respectively. Normal operating hot leg and cold leg temperatures are given as 583°F and 537°F, respectively.

The PNP UFSAR (Reference 3) Section 4.7 describes PCS leak detection equipment available and operator actions taken to support PCS leak detection. This information is repeated below:

Leak Detection

Small leaks from the Primary Coolant System can be detected by one or a combination of the following systems:

1. Containment Atmosphere Relative Humidity - Each of 4 humidity detectors (1 in each steam generator compartment, 1 at the refueling floor near the reactor refueling cavity and 1 near the containment dome) is capable of detecting a change of humidity of 10% which would result from approximately 150 gallons of primary water leakage.

2. Containment Sump Level - Containment sump water level indication is provided in the main control room by level indicators which can be used to detect primary coolant system leakage. One level indicator actuates a high-high-water level alarm at 12 inches.

3. Containment Area Radiation - One radiation monitor, sensing from the discharge of all operating containment air coolers, is capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel.

4. Reactor Vessel Flange Leak Off - The inner seal leakage goes to a closed drain line and leakage will be detected by a pressure alarm set at 1,500 psig which will be activated by a steam leakage from the reactor of approximately 130 in^3. The outer seal liquid leakage is collected and drained to a closed drain line and will be detected by action of a level switch set at 120 inches which will result from a liquid accumulation of approximately 35 in^3.

5. Steam Generator Tube Leakage - Radiation detectors are provided to monitor the liquid effluent from the blowdown tank and gas effluents from the air ejector. The monitors have a sensitivity of 4 x 10-6 µCi/cm3 and can be set to alarm at 1.0 x 10-5 µCi/cm3 depending on normal background. The expected background will require that the alarm point be set higher than 1.0 x 10-5 µCi/cm3 but will be well below the activity released by a 5 gpm primary to secondary tube leak with 1% failed fuel.

6. A leak between the Component Cooling Water System and the Primary Coolant System via the primary coolant pump seals can be detected by a high component cooling water system surge tank level alarm and a high component cooling water system radiation alarm.

7. Each control rod drive mechanism face seal is equipped with a leak off which is piped to the floor drains leading to containment sump. Each leak off contains a thermocouple which will activate an alarm should above-normal temperatures occur.

8. The safety and power-operated relief valves may be a potential source of contained leakage. Seat leakage of these valves drains to the pressurizer quench tank and excess leakage would be detected by temperature monitors located in the valve discharge piping. Large amounts of seat leakage would also be detected by increases in level and temperature in the pressurizer quench tank. In 1979, pursuant to NUREG-0578, acoustical monitors were added to these valves to provide positive position indication in

the control room.

9. Small leaks may also be determined by comparing charging pump and letdown flow rates and observing changes in pressurizer level.

10. Primary Coolant System Uncontrolled Bleed-off to T-74 (Primary System Drain Tank) – Leak rate from this header can be collected and measured following alignment of an isolation and sample valve.

Operator Action Following Leak Detection

In the event a small leak is indicated in the Primary Coolant System, immediate steps will be initiated to identify the source and isolate the leak if possible.

The initial operator action following an indication of a leak in the Primary Coolant System is to check pressurizer level and the Chemical and Volume Control system response. The next step is to attempt to determine the leak rate. This may be done by comparing charging and letdown flows and observing pressurizer level.

TS address limits for operating with identified and unidentified PCS leakage. If the leakage rate exceeds the ability of the Chemical and Volume Control System to maintain pressurizer level, the reactor is manually tripped.

2.2 Technical Specifications Requirements

A reliable leak detection system is required for application of the LBB methodology. This reliability is necessary to monitor initiation of a leak in the reactor coolant pressure boundary so that appropriate actions can be taken to place the plant in a safe condition.

At PNP, two TSs address PCS leakage and leakage detection (Reference 1):

TS 3.4.13 PCS Operational LEAKAGE – This TS limits Primary Coolant Pressure Boundary leakage from unidentified and identified sources and primary to secondary leakage through any one Steam Generator (SG).

TS 3.4.15 PCS Leakage Detection Instrumentation – This TS requires operability of three of the following PCS leakage detection instrumentation channels:

a. One containment sump level indicating channel;

b. One containment atmosphere gaseous activity monitoring channel;

c. One containment air cooler condensate level switch channel;

d. One containment atmosphere humidity monitoring channel.

These PNP TS require that the reactor coolant leakage detection instrumentation be operable in operating Modes 1 through 4. The PNP primary coolant leakage detection instrumentation is consistent with the intent of the regulatory positions in RG 1.45 (Reference 5) in that they provide the means for detecting, and to the extent practical, identifying the location of the source of reactor coolant leakage. In addition, the primary coolant leakage detection instrumentation assures that at least one method is capable of detecting a 1 gpm leak in 1 hour of unidentified leakage.

2.3 Reason for the Proposed Change

Holtec intends to credit the work performed by the CEOG leak-before-break evaluation of the primary coolant loop piping (Reference 2) to eliminate the dynamic effects of postulated primary loop pipe ruptures from the PNP licensing basis. To support the planned core shroud bolt inspection process, the reduced Loss of Coolant Accident (LOCA) loads, a benefit of the LBB analysis, will correspond to a reduced required number of core shroud bolts, minimizing the potential need of a core shroud bolt replacement effort, and minimizing personnel radiation exposure and resource expenditures.

Additionally, with the approval to eliminate PCS pipe break loads, the requirement to design against a double guillotine break would translate to additional margin in future piping analysis. Holtec desires the additional margin in the piping, pipe support, and component support analyses to eliminate future needs for additional pipe restraints or supports whose installation and maintenance could require personnel radiation exposure as well as resource expenditures to engineer, install, and maintain.

The implementation of LBB requires a license amendment under 10 CFR 50.90, *Application of Amendment of License, Construction permit, or Early Site Permit*, because one or more of the criteria of 10 CFR 50.59(c)(2) applies to LBB.

2.4 Description of the Proposed Change

Holtec requests NRC approval to apply LBB methodology to the PCS piping at PNP. The proposed change will allow the elimination of large break asymmetric dynamic loads in the PCS hot leg and cold leg piping from the licensing basis. The dynamic loads due to breaks in all attached piping will still be included in the licensing basis. There are no TS that need to be revised for this change. However, there will be changes made to the TS Bases and the UFSAR through the 10 CFR 50.59 review process.

Changes will be made to the following UFSAR Sections to reference the CEN-367-A (Reference 2) approved report and changes to the design loads to the Reactor Vessel Internals (RVIs):

- Section 4.7 “Primary Coolant Pressure Boundary Leakage Detection.”
- Section 14.17.3 “Reactor Internals Structural Behavior Following a LOCA.”

In addition, the reference to the LAR approval and reference to RG 1.45 will be added to PNP TS 3.4.13 and 3.4.15 Bases.

Background

In a nuclear power plant, structures, systems, and components important to safety require protection from accidents, including pipe breaks. A pipe break creates dynamic forces due to fluid discharge and pipe whip as a reaction to the jet created at the break location. The magnitude of the dynamic forces generated by a pipe break depends on the size of the break. One method to determine the size of the break is to assume an instantaneous formation of an arbitrary break and separation across the pipe diameter. This deterministic postulation is non-mechanistic and provides the severest condition requiring a complex protection system to counteract the dynamic forces created by the pipe break.

In reality, a pipe break occurs through the formation of a tiny crack in the line that, if unstable, develops into a full-size crack over time. A second method for estimating a break makes use of this fact by examining the potential for, and the duration of the crack formation. Through this analysis, it is possible to predict whether a crack will form and, in the event of its formation, whether sufficient warning will be available to safely shut down the plant. This complex analysis requires reliable engineering data of the pipe material, its configuration and plant operating experience. However, a successful implementation of this methodology reduces the complexity of systems required to protect the plant against pipe breaks. The application of this methodology, referred to as LBB methodology, reduces radiation exposure and maintenance costs while maintaining plant safety.

The use of LBB analysis is allowed by 10 CFR 50, Appendix A, GDC 4, when reviewed and approved by the NRC, to eliminate from the design basis the dynamic effects of the pipe ruptures postulated in nuclear power plant units. An NRC staff-approved LBB analysis permits licensees to remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports and their internals, and perform other related changes in operating plants.

3 TECHNICAL EVALUATION

3.1 Technical Assessment

The technical assessment of the application of LBB methodology to the PNP Licensing Basis is discussed below by addressing 1) The base CEOG LBB evaluation in Reference 2, 2) The application of the CEOG LBB evaluation to PNP (including the changes made to the plant since the CEOG analysis approval), and 3) The PNP adherence to RG 1.45 (Reference 17).

3.1.1 CEN-367-A CEOG LBB Evaluation

CEN-367-A (Reference 2) is an LBB analysis performed in accordance with 10 CFR 50, Appendix A, GDC 4, NUREG-1061, *Evaluation of Potential for Pipe Breaks*, Volume 3. The analysis performs an evaluation of the CEOG plants by evaluating LBB in the hot leg and cold leg PCS piping.

This evaluation was approved for use at PNP and other CEOG plants on October 30, 1990. The evaluation approach and findings are discussed in Reference 2. As stated in Reference 2, Appendix A, “Tables 1 through 9 demonstrate that the PNP primary piping system is enveloped for piping loads within their grouping, and the conclusions of the generic LBB evaluation are applicable.”

The staff’s evaluation of GDC 4 compliance in the Nuclear Regulatory Commission (NRC) Safety Evaluation (SE) included the following “points”:

(1) Normal operating loads, including pressure, deadweight, and thermal expansion, were used to determine leak rate and leakage-size flaws. The flaw stability analyses performed to assess margins against pipe rupture at postulated faulted load conditions were based on normal plus SSE loads, in the stability analysis, the individual normal and seismic loads were summed to maximize the postulated flaw opening area. Leak-before-break evaluations were performed for the limiting location in the piping.

(2) For CEOG facilities, there is no history of cracking failure in reactor coolant system (RCS) primary loop piping. The RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle).

3) The material tensile and fracture toughness properties were provided. Because the safe ends on the subject CEOG primary loop piping consist of cast stainless steel, the thermal aging toughness properties of cast stainless steel materials were considered based on data from NRC sponsored research at the Argonne National Laboratory. The fracture toughness for ferritic steel was estimated based on data from NRC sponsored research at the Battelle's Columbus Division, and was consistent with that used in the development of

ASME Code Case N-463. The material tensile properties were based on testing typical CEOG piping material.

(4) CEOG contended that CEOG plants have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 such that a leakage of one gallon per minute (gpm) in one hour can be detected. In estimating leakage through a postulated flaw, the CEOG used a constant factor of 250 gpm/in of flaw opening area. The staff does not accept this approach because leakage is a complicated thermal-hydraulic phenomenon. However, based on staff's calculations using the PICEP computer code, the staff found the CEOG results acceptable for this case. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems; the margin is a factor of 10 in leakage and is consistent with the guidelines of NUREG-1061, Volume 3 (November 1984). The staff did not evaluate the subject CEOG plants according to the guidance in Regulatory Guide 1.45 in this review. This will be addressed on a plant specific basis when licensees submit requests for LBB applications.

(5) In the flaw stability analyses, the staff evaluated the margin in terms of load for the leakage-size flaw under normal plus SSE loads. The results of the CEOG analyses, consistent with the results of the staff's calculations, indicated the margin exceeded 1.4 when the individual normal and seismic loads were summed. The margin is consistent with the guidelines of NUREG-1061, Volume 3 (November 1984).

(6) Similar to item (5) above, the margin between the leakage-size flaw and the critical-size flaw was also evaluated in the flow stability analyses. The results of the CEOG analyses, consistent with the results of the staff's calculations, indicated the margin in terms of flaw size exceeded 2. The margin is consistent with the guidelines of NUREG-1061, Volume 3 (November 1984).

As a condition of approval in the Nuclear Regulatory Commission (NRC) Safety Evaluation (SE), the NRC stated:

> “However, when referencing this CEOG topical report as a technical basis for applying LBB to primary loop piping, licensees must submit information to demonstrate that leakage detection systems installed at the specific facility are consistent with Regulatory Guide 1.45.”

3.1.2 CEN-367-A Application to PNP

The SE for the CEN-367-A analysis concluded: “Thus, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of the subject CEOG plants is sufficiently low such that dynamic effects associated with postulated pipe breaks need not be a design basis”.

The NRC letter approving CEN-367-A includes the statement: “We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved.” Thus, the evaluation herein focusses on applicability to Palisades.

Palisades is one of the listed plants that the CEN-367-A analysis was applicable to and bounding for.

Since the SE was written in October of 1990, PNP underwent two changes which could have impacted the operation of the plant, and thus impacted the plant condition assumptions in the Reference 2 analysis. These are assessed below:

Measurement Uncertainty Power Uprate

PNP underwent a small power uprate (2530 MW thermal to 2565.4 MW thermal) on June 23, 2004, as documented in TS amendment 216. However, as noted in Section 14.1.1 of the UFSAR, the uprate did not result in an actual change to the licensed power level, since it only credited a lower analysis uncertainty and the actual analyzed power level didn’t change. Therefore, this plant change had no impact on plant conditions.

Steam Generator Replacement

In addition, the plant replaced Steam Generators (SGs) in 1990 as discussed in the updated FSAR, Section 4.3.4.2. The evaluation below addresses the impact of the replacement SGs on the applicability of CEN-367-A for Palisades. To determine if the Palisades steam generator replacement has any impact on the applicability of CEN-367-A to Palisades, the steam generator replacement is examined based on Materials, and Loads.

Materials

The original SGs were designed and fabricated to the requirements of the 1965 edition of the American Society of Mechanical Engineers (ASME) Code, Section III, *Rules for Construction of Nuclear Facility Components*, including all addenda through Winter 1965. The replacement SGs were fabricated to the requirements of the 1977 Edition of the ASME Code. The stress report for the replacement SGs is based on the 1977 Edition of the ASME Code. Based on a comparison of the requirements included in the

1977 ASME Code against those specified in the 1965 edition of the Code:

- The design of the replacement SGs is consistent with the PCS.
- Welding of the PCS replacement elbows to the new SGs and PCS piping was done using the gas tungsten arc welding (GTAW) process and meets the requirements of ASME Section IX; *Welding, Brazing, and Fusing Qualifications*, using appropriate welding filler material. The GTAW weld process generally produces better fracture properties than the submerged arc welding (SAW) process. Since CEN-367-A used lower bounding fracture properties based on the SAW weld process, it is determined that the material properties used in CEN-367-A are acceptable.

The replacement SGs at PNP were fabricated to a more recent version of the ASME Code (1977 Edition), but the materials and welding processes used are compatible with the original CEN-367-A analysis. The GTAW welding process used for the replacement results in improved fracture properties, and since CEN-367-A assumed a conservative lower bound for fracture properties (based on the SAW process), the material properties remain acceptable. Therefore, the replacement steam generators do not impact the applicability of CEN-367-A for Leak-Before-Break (LBB) analysis at PNP.

Loads

Comparison of the original and replacement SGs shows that the nominal full-load operating weight of the replacement SGs is approximately 1 percent higher than the original SGs, and the associated center of gravity is about 9 inches lower.

The resulting changes are negligibly small, and their effects on piping loads, SG support loads, and the SG dynamic characteristics, are not significant. The original SG bottom support skirt assembly and sliding base support were reused for the replacement SGs without modification. The effects of the small changes in SG weight, center-of-gravity, the vibration frequency on the SG supports and connected systems have been evaluated and determined to be insignificant. Stress analyses were updated for main feedwater, auxiliary feedwater, and main steam piping, but there is no record of changes to PCS piping or supporting structures. In addition, the design cyclic loading and static stress design limits of the replacement SGs are equal to or more conservative than similar limits for the original SGs. The replacement program, including design, fabrication and installation, meets ASME requirements.

Based on these findings the replacement SGs at PNP have no impact on the plant primary loop piping and the LBB analysis. The small increases in weight and changes to the center of gravity are negligible and do not affect the hot leg and cold leg loads considered in CEN-367-A. The updated stress analyses confirm that the changes do not invalidate the original LBB analysis, and the design limits for the replacement SGs are at least as conservative as those of the original SGs. Therefore, CEN-367-A remains applicable for PNP, and the replacement SGs do not require any changes to the LBB analysis.

The following information is provided to assess the current condition of the PCS hot and cold leg piping in light of any impact it may have on the assumptions made in Reference 2, thus assuring that the results of the analysis are still applicable to PNP.

Evaluation of PCS Piping Aging Impact on LBB

Since the approval of CEN-367-A, PNP has committed to manage plant material aging issues by implementing an Aging Management Program (AMP) (Reference 3, Section 1.9.1). The AMP for the PCS maintains adherence to safety and regulation by managing aging effects, anticipating future updates as knowledge and regulations evolve, and it ensures that all changes are made within the framework of established regulatory processes. Pertinent to the LBB application, PNP implements the following plant programs:

1) Inservice Inspection (ISI) Program - The ISI program is discussed in UFSAR, Section 1.6. The ISI program at PNP assures that degradation of PCS piping will not impact the analysis assumptions.
2) PNP Life Extension and Time Limited Aging Analysis (TLAA) - The TLAA program is described in UFSAR, Section 1.9. The fatigue analysis of the PCS piping is of particular interest.
3) Nickel Alloy Program - Primary Water Stress Crack Corrosion (PWSCC) is discussed in Section 1.9 of the PNP UFSAR. This program manages PWSCC in PCS piping at PNP.

Below is a summary of each program.

Inservice Inspection Program

The ASME Section XI Inservice Inspection Program (ISI) at PNP (UFSAR, Section 1.6) is designed to facilitate inspections and assessments to identify and correct degradation in key piping, components, and supports within the plant's nuclear systems. It applies to Class 1, 2, and 3 components, which are critical for ensuring the structural integrity and safe operation of the plant. This program helps to identify degradation early and take corrective actions to maintain the integrity of the plant’s key pressure-retaining components, ensuring that they continue to meet safety standards throughout the period of extended operation.

The ISI program is in the fifth 10-year ISI period, which began in December 2015. The ISI program maintained safety and compliance with evolving industry standards by program adherence to ongoing updates and regulation. The ASME Section XI ISI Program at PNP is a comprehensive approach to monitor and maintain the integrity of critical nuclear plant components, ensuring continued safe operation and compliance with regulatory standards. This program involves a variety of inspection techniques and covers numerous components essential to the plant’s safety. The ISI program at PNP is implemented through Reference 6.

A manual search through outage inspection reports in the PNP database yielded no indications of flaws or leakage in the PCS piping throughout the life of the plant.

PNP Life Extension and TLAA

Upon approval of License Renewal in 2007 (Reference 11), Time-Limiting Aging Analysis (TLAA) was implemented at PNP. The PNP UFSAR, Section 1.9, discusses the programs and activities credited for managing the effects of aging during the period of extended operation. These materials management programs at PNP assure that the CEOG LBB analysis (Reference 2) remains applicable to PNP.

The PCS piping metal fatigue analysis is addressed in Section 1.9.2.2 of the UFSAR, item g, “The ASME III Class A Primary Coolant Piping Fatigue Analyses.” Fatigue in the PCS piping was calculated in certain components that were selected based on their susceptibility to cyclic stresses and their importance in maintaining the integrity of the coolant system. Utilizing a conservative approach to estimate stress ranges and fatigue cumulative usage factor (CUF), the HL CUF is 0.07551 and the CL CUF is 0.7531, which is still below the acceptable CUF of 1.0. Additionally, UFSAR Section 1.9.1.26 describes the implementation of the Fatigue

Monitoring Program, which is a proactive measure to ensure safe continued operation by addressing the risks associated with metal fatigue by managing and tracking metal fatigue in components of the PCS pressure boundary during the period of extended operation.

The materials of primary interest for the LBB analysis are the Cast Stainless Steel (CSS) safe ends because of the sensitivity to thermal aging at the SA-516 Grade 70 weld locations. Based on CEN-367-A, Section 2.3, Item 3, the material tensile and fracture toughness properties were provided to the Staff. Because the safe ends on the subject CEOG primary loop piping consist of CSS, the thermal aging toughness properties of CSS materials were considered based on data from NRC sponsored research at the Argonne National Laboratory (ANL). The fracture toughness for ferritic steel was estimated based on data from NRC sponsored research at the Battelle's Columbus Division, and was consistent with that used in the development of ASME Code Case N-463. The material tensile properties were based on testing typical CEOG piping material.

Primary Water Stress Corrosion Cracking (PWSCC) Concerns

PWSCC of Alloy 600, including Inconel 82/182 welds, in the PCS is addressed by the Nickel Alloy Program (UFSAR, Section 1.9.1.1), which aims to actively monitor and manage the integrity of critical Nickel Alloy components in the PCS to prevent aging-related failures and ensure the plant's continued safe operation. The program is structured around several key activities:

1. PWSCC Susceptibility Assessment: The program utilizes industry models to assess which components in the PCS are susceptible to PWSCC. This helps identify those areas that require additional focus.
2. Primary Coolant Chemistry Control: The program emphasizes the importance of managing the primary coolant chemistry to mitigate the risk of PWSCC. Proper chemistry is essential for minimizing stress corrosion cracking.
3. In-Service Inspections (ISI): The program includes regular inspections of critical components such as pressurizer penetrations, reactor vessel head penetrations, and Alloy 82/182 PCS pressure boundary welds. These inspections follow the guidelines of the ASME Boiler and Pressure Vessel Code (BPV Code), specifically Section XI, which sets the ISI rules for nuclear power plant components. The applicable table for these inspections is IWB-2500-1.
4. Augmented Inspections and Preemptive Repairs/Replacement: For components or welds identified as susceptible to PWSCC, the program may include augmented inspections or even preemptive repair or

replacement to prevent failures and maintain the integrity of the PCS pressure boundary.

PWRSCC-susceptible materials at PNP are mitigated through implementation of a Nickel-Alloy program (Reference 23).

Based on the Plant Programs described above, the PNP PCS piping under LBB consideration is not prone to fatigue or corrosion. Material degradation is managed to eliminate the risks and concerns of age-related material degradation.

3.1.3 Compliance with Regulatory Guide 1.45

As noted in Section 3.1.1, it was a condition of application of the CEOG LBB analysis in CEN-367-A (Reference 2 of this enclosure), that each plant referencing CEN-367-A to justify LBB must demonstrate that leakage detection systems installed at the specific facility are consistent with RG 1.45, Revision 0.

3.1.3.1 ***Background***

A review of PNP records and the NRC's online document retrieval system in the Agencywide Documents Access and Management System (ADAMS) was performed. The earliest identified PNP references to RG 1.45 Revision 0 (Reference 5) were associated with the PNP Integrated Plant Safety Assessment System Evaluation Program (SEP) which resulted in Reference 12.

In Reference 12, NRC documented that PNP would require a TS change in order to meet the regulatory positions in RG 1.45 (1973) as presented in SEP Topic V-5 and that action could be deferred until related SEP Topic III-5.A (Effects of Pipe Break on Systems, Structures and Components Inside Containment) final actions were identified. The NRC acknowledged that development of TS changes concerning the operability of leakage detection systems would depend on the outcome of Topic III-5.A, and would be the only required action necessary associated with SEP Topic V-5 (Reference 12, Section 4.15.2 and Table 4.1).

On 12/13/1983 (Reference 13) PNP submitted a Technical Specification Change Request (TSCR) to propose adding new leakage detection systems and surveillance requirements which would resolve outstanding SEP Topic V-5 from Reference 3, Section 14.5.2. The TSCR was later replaced with a TSCR submitted on 5/23/1985 (Reference 14) which also referenced it was being submitted in accordance with Reference 3 section 4.15.2. In turn, this TSCR was later replaced with a TSCR on 11/15/1991 (Reference 15) which combined several instrumentation and controls related changes into one submittal. The Reference 15 TSCR provided the PCS Leakage Detection operability and surveillance requirements needed to close SEP Topic V-5 and was approved on 10/26/94 (Reference 16) as PNP License Amendment 162.

The addition of PNP PCS Leakage Detection TS requirements for Containment

Sump Level, Atmosphere Gas Monitor, Humidity Monitor and Air Cooler Condensate Flow Switch as part of Amendment 162 to the PNP License closed SEP Topic V-5, resulted in PNP becoming consistent (or compliant) with RG 1.45 1973.

3.1.3.2 Compliance with RG 1.45 at PNP

As required by the NRC SER for CEN-367-A, PNP is providing the following information to demonstrate that the PCS leakage detection systems installed at the PNP are consistent with the regulatory positions contained in NRC Regulatory Guide (RG) 1.45, *Guidance on Monitoring and Responding to Reactor Coolant System Leakage*, Revision 1, dated May 2008 (Reference 18). A comparison of the diversity and sensitivity of Palisades' systems to the regulatory positions specified in Section C of RG 1.45 is presented. The section headers are in **Bold** and the sub-section regulatory positions in *italics* taken from RG 1.45 Revision 1.

The TS references in this LAR are based on the Power Operation Technical Specifications (POTS) submitted with Reference 1. The Updated Final Safety Analysis Report (UFSAR) references in this LAR are based on UFSAR revision 35 (Reference 3). Changes made to the UFSAR after Revision 35 related to this amendment will be evaluated as required by 10 CFR 50.59. Plant procedures referenced by this LAR are to those versions applicable to power operations after PNP transitions to a Power Operations Licensing Basis (POLB) per Reference 1.

1. General Positions

1.1. *The source and location of reactor coolant leakage should be identifiable to the extent practical, and the plant should measure the leakage rate.*

TS 1.1 establishes the definitions of Identified, Unidentified and Pressure Boundary LEAKAGE for the Palisades Renewed Facility Operating License (RFOL).

TS 3.4.13 provides the Primary Coolant System (PCS) Operational LEAKAGE Limiting Condition of Operation (LCO) for pressure boundary LEAKAGE (none), unidentified LEAKAGE (1 gpm), identified LEAKAGE (10 gpm) and primary to secondary LEAKAGE (150 gallons per day) through any one steam generator (SG).

Surveillance Requirement (SR) 3.4.13.1 requires Operators to verify operational LEAKAGE is within limits by performance of PCS water inventory balance. SR 3.14.13.2 requires operators to verify primary to secondary LEAKAGE is </= 150 gallons per day through any one SG.

TS 3.4.15 provides the PCS Leakage Detection Instrumentation channel LCO and associated SR.

In 2010 PNP implemented Administrative Procedure (ADMIN) 4.19 “PCS Leak Rate Monitoring Program.” The purpose of this procedure is to specify requirements for assessment and response to a rise in PCS Unidentified Leakage. This procedure describes administration, assessment and response guidelines, data verification guidelines, and actions in response to a rise in PCS Unidentified Leakage. ADMIN 4.19 follows industry best practices per Westinghouse PWR Owners Group Letter OG-07-286 (Reference 19).

1.2. *The plant should collect or otherwise isolate leakage to the primary reactor containment from identified sources so that the following criteria are fulfilled:*

(i) Flow rates from identified sources are monitored separately from the flow rates from unidentified sources.

PNP utilizes TS Surveillance Procedure DWO-1 “Operator’s Daily/Weekly Items Modes 1, 2, 3 and 4” to demonstrate compliance with TS 3.4.13, SR 3.4.13.1, and LCO 3.4.15, Condition A. “One or two required leak detection instrument channels inoperable.” DWO-1 establishes primary system leakage calculation precautions and limitations and provides explicit direction as to how to perform a primary system leakage calculation. The procedure includes steps to calculate total, identified and unidentified PCS leakage. Unidentified leakage and identified leakage are determined by performance of a PCS water inventory balance (mass). The total leakage calculation accounts for system pressure, temperature and level variations during the test as well as known makeup additions and leakage sources outside the PCPB (e.g., the Chemical Volume and Control (CVCS) system).

Identified (monitored) PCS leakage sources include:

• Control Rod Drive Mechanism (CRDM) seals.

• Primary Coolant Pump (PCP) seal uncontrolled bleed off.

• Other positively identified and quantified primary coolant leakage sources such as Reactor Coolant Pressure Isolation Valve (PIV) leakage, fitting or gasket leaks from the PCPB that are collected and measured.

(ii) The plant can establish and monitor the total flow rate.

Total flow rate is established as described in the response to position 1.2 (i). Total flow rate is monitored as required by TS SR 3.4.13.1, DWO-1 and ADMIN 4.19.

1.3. The plant should monitor critical components of the RCPB for leaks.

In addition to the components utilized in the DWO-1 leak rate calculation, ADMIN 4.19 Attachment 4 lists parameters to be monitored for unidentified PCS leakage.

1.4. The plant should monitor intersystem leakage for systems connected to the RCPB.

Intersystem leakage monitoring has been incorporated in the system design of the following piping systems, which directly interface with the PCPB.

(i) Main Steam System

A primary-to-secondary leak would occur through the SGs to the main steam system. PNP TS 3.4.13.d provides a limit on primary-to-secondary leakage. There are five monitors available to identify this leakage:

- one main condenser air discharge system (off-gas) monitor RIA-0631,
- two main steam line radiation monitors RIA-2323, 2324,
- one SG sample cooler/blowdown system monitor RIA-0707,
- one SG blowdown tank vent monitor RIA-2320,

Procedure CH 3.31 “Primary to Secondary Leak Rate Determination” details the determination of primary to secondary leak rate within the SGs via main condenser off-gas sampling and analysis (Modes 1 and 2), SG tritium sampling (Modes 1 through 4), and using RIA-0631 (Modes 1 and 2). RIA-0631 is monitored by Control Room Operators hourly with a resolution of 0.01 counts per minute (cpm) (Primary Plant Computer (PPC)).

RIA-0707 is monitored by Control Room Operators hourly with a resolution of 0.01 cpm (PPC).

(ii) Component Cooling Water (CCW) System

A leak between the CCW system and the PCS via the primary coolant pump seals can be detected by a high CCW system surge tank level alarm and a high CCW system radiation alarm (Reference 3, Sec 4.7.1.6). The CCW system is equipped with a radiation monitor (RE/RIA-0915) mounted on the CCW pump discharge piping. The monitor can detect PCS leakage into the CCW system (Reference 3, Table 11-15). In addition, PCS leakage into the CCW system would be reflected in a CCW surge tank water level rise and CCW water temperature change. CCW surge tank water level (LT/LIA-0920) is trended and logged in the control room hourly. Temperature instruments are installed on the piping sections of the return lines from the primary coolant pump heat exchangers. Signals from these temperature instruments are sent to recorders in the control room.

(iii) Shutdown Cooling (SDC) System

PNP TS 3.4.14 addresses the limits on PCS pressure isolation valve (PIV) leakage. PNP TS 3.4.14 also requires that both of the SDC suction valve interlocks to be operable when operating in Modes 1 through 3. Any PCS leakage through the interlocked valves is directed via a pressure relief valve, to the primary quench tank, which would cause quench tank water level and temperature to rise. The quench tank water level (LIA-0116), pressure (PIA-0116) and temperature (TIA-0116) are indicated and alarm in the control room.

(iv) Safety Injection System (SIS)

PCS leakage into the SIS is also subject to PNP TS 3.4.14 limits on PIV leakage. The PCS pressure boundary consists of two isolation valves at each boundary interface. PCS leakage through the first PIV would result in a pressure increase in the piping section between the two isolation valves or overflow to the safety injection tanks (SITs). The control room monitors the PIV leakage by the pressure instrumentation on the piping sections and the level instruments on the SITs.

(v) Chemical Volume Control System (CVCS)

The CVCS provides continuous makeup flow to, and letdown flow from, the PCS and is a directly connected system. Flow during normal power operations is provided by a single variable speed charging pump, with two fixed speed charging pumps in standby. Letdown flows through one, two or three letdown orifices which restrict flow to the less than the makeup capability of the charging pumps. Charging and letdown flow trends are monitored hourly by Control Room Operators with a resolution accuracy of 0.01 gpm using the PPC. Variations in charging flow would provide early indication of small PCS leaks.

1.5. The capabilities of the leakage monitoring systems should be known. In addition, the capabilities should ensure effective management of leakage.

The capabilities of the PNP leakage monitoring system is described in the responses to the regulatory positions provided in this enclosure. The extent to which the various components of the leakage monitoring systems are utilized is in accordance with the Plant TS, Admin 4.19, and supported by the Surveillance and Operating procedures.

While some aspects of RG 1.45 regulatory positions are not individually met by the PNP PCS Leak Rate Monitoring Program, as a collective the program meets or exceeds the intent of RG 1.45 through diversity of indication, procedure quality and demonstrated continuous improvement.

Tables 1 through 4 in the Attachment to this enclosure provide a summary of the core components of the PNP PCS Leak Rate Monitoring Program. The purpose of the Attachment is three-fold:

- Provide a line of sight from the original NRC evaluation of PNP leakage detection systems to the present time, based on the format the NRC utilized in Reference 10. Table 4 lists components not considered in Reference 10, some of which did not exist when the original evaluation was performed.

- Demonstrate that the PNP leakage monitoring system has evolved since the original NRC evaluation against RG 1.45 Revision 0 in 1982.

- Demonstrate that the capabilities of the system are known, and all components are tied to Plant Operating Procedures.

2. Leakage-Monitoring-Related Positions

2.1. *Plant procedures should include the collection of leakage to the primary reactor containment from unidentified sources so that the total flow rate can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gal/min (0.19 L/min).*

Primary coolant leakage reaching the containment building sump would be found by the operator's periodic surveillance of the containment level indicators or annunciated in the control room by activation of the sump high-level or high-high alarms (UFSAR 5.1.5.1.5). Operators refer to Plant TS 3.4.13, and procedures as described in the responses to positions 1.1 and 1.2 (above) to differentiate between unidentified leakage and total flow rate.

The suggested total flow rate >/= 0.05 gpm regulatory position was not in existence during PNP initial design, when RG 1.45 R0 was issued, or during the SEP evaluation of the PNP design against the RG 1.45 R0 regulatory positions. However, PNP inventory balance leak rate calculation capability has evolved such that either by manual (hand calculation) or automated PPC means; total, identified and unidentified leak rate are calculated and recorded to the nearest 0.001 gpm which is well below the >/= 0.05 gpm guideline flow rate. Control Room Operators monitor a 15-minute time-average unidentified PCS leak rate trend hourly using the PPC with a resolution of 0.01 gpm.

Containment sump level (LT-0382) and sump fill rate trends (PPC) are typically displayed continuously by Control Room Operators using the PPC and are required to be monitored a minimum of hourly as part of Control Room Operator rounds. The PPC trend resolution for sump level and sump fill rate are 0.01 Feet and 0.01 gpm, respectively. PCS leak rates of </= 1 gpm are readily identifiable using these trends and can be corroborated against the PPC unidentified PCS Leak Rate 15-minute time averaged calculation trend, which also has a resolution of 0.01 gpm.

PPC calculations for inventory mass balance and unidentified leak rate implementations are covered by the PNP software quality assurance program. The PPC inventory mass balance calculation is not safety related but is augmented quality and considered a Critical Digital Asset.

As stated in the response to Regulatory Position 1.4, CVCS charging flow is monitored hourly by Control Room Operators with a resolution accuracy of 0.01 gpm using the PPC. A rising trend in charging flow could be direct indication of total leakage. Monitoring charging flow provides for diversity in capability for identifying PCS leakage.

2.2. *The plant should use leakage detection systems with a response time (not including the transport delay time) of no greater than 1 hour for a leakage rate of 1 gal/min (3.8 L/min).*

The instrumentation identified in the response to regulatory positions 2.1 and 2.3 is permanently installed plant instrumentation with indication, alarm, trending and response time capabilities as summarized in the Attachment to this enclosure.

2.3. *Plant technical specifications should identify at least two independent and diverse instruments and/or methods that have the detection and monitoring capabilities detailed above. The methods to consider for incorporation in the technical specifications include, but are not limited to, the following:*

(i) monitoring sump level or flow,

PNP TS 3.4.15.a provides the LCO and SR associated with monitoring sump level. Containment sump water level indication is provided in the main control room by level indicators which can be used to detect PCS leakage (Reference 3, Section 4.7.1.2). Containment sump level indication provides indication of leakage into the sump from sources such as the PCS and service water. There are two level switches (LS-0358/0360) and one level indicator (LT/LIA-0358) to monitor the containment sump water level. The level switches activate a single high-level alarm. The level indicator activates a separate high-level alarm. Level indication is provided in the control room and all alarms are annunciated in the control room. This original instrumentation was not qualified for harsh environments; therefore, two additional level transmitters (LT-0382/0383) were added in 1982, pursuant to NUREG-0578, to provide diverse, redundant and environmentally qualified sump level indication.

Two level transmitters (LT-0382 and LT-0383) provide sump level indication to two control room recorders (LPIR-0382 and LPIR-0383). One of these instruments is required to be operable per TS 3.4.15. Operability is verified each shift by Control Room Operators. Containment sump level (LT-0382) and sump fill rate trends have a PPC trend resolution of 0.01 Feet and 0.01 gpm, respectively.

LS-0358 and LS-0360 actuate a control room containment sump high level alarm at 4 inches from the bottom of the sump, elevation 585.33ft.

LT/LIA-0359 actuates a control room containment sump high-high level alarm at 12 inches or 10% from the bottom of the sump. LIA-0359 is utilized for hourly trending as a backup to LT-0382 in the event PPC trending is not available. A 293-gallon liquid addition to the sump results in a 0.1 foot of rise in the sump and corresponds to a 1% change on LIA-0359 or 29.3 gal per 0.01 ft.

Operator action for either the high or high-high sump level alarm is directed through plant procedures.

(ii) monitoring airborne particulate radioactivity, and

PNP TS 3.4.15 provides the LCO and SR associated with containment atmosphere gaseous activity monitoring. PNP does not have an airborne particulate radiation monitor in containment, rather, there are seven separate airborne gaseous radiation monitors installed in containment which aid the operators with PCS leak detection. The capability of the containment gaseous radiation monitors is included in the response to RG 1.45 Regulatory Position 2.3.a of this enclosure.

The lack of an airborne particulate radiation monitor in containment was originally identified as part of the NRC review of the PNP design with respect to RG 1.45 during the SEP evaluation (Reference 12).

In Reference 10, during review of SEP Topic V-5, NRC identified that PNP did not meet this requirement and presented modifications to be considered during the integrated safety assessment. In response, PNP proposed deferring action until Topic III-5.A (Effects of Pipe Break on Systems, Structures and Components Inside Containment) final actions were identified. NRC agreed that development of TS changes concerning the operability of leakage detection systems would depend on the outcome of Topic III-5.A and would be the only required action necessary associated with SEP Topic V-5 (Reference 12, Section 4.15.2 and Table 4.1).

In Section 4.15 of Reference 12, the NRC states,

"Although all of the leakage-detection systems recommended by the Regulatory Guide are not present, PNP incorporates six additional diverse systems. Taking all of these systems into consideration, it is the staff's judgment that a 1-gpm leak from the RCPB to the containment will most probably be detected within 24 hours."

On December 13, 1983 (Reference 13) PNP submitted a Technical Specification Change Request (TSCR) to propose adding new leakage detection systems and surveillance requirements which would resolve outstanding SEP Topic V-5 from

Reference 12, Section 14.5.2. The TSCR was later replaced with a TSCR submitted on 5/23/1985 (Reference 14) which also stated it was being submitted in accordance with Reference 12 section 4.15.2. In turn, this TSCR was later replaced with a TSCR on 11/15/1991 (Reference 15) which combined several instrumentation and controls related changes into one submittal. The Reference 15 TSCR provided the PCS Leakage Detection operability and surveillance requirements needed to close SEP Topic V-5 and was approved on 10/26/94 (Reference 16) as PNP License Amendment 162.

The PNP PCS Leakage Detection TS requirements for Containment Sump Level, Atmosphere Gas Monitor, Humidity Monitor and Air Cooler Condensate Flow Switch were added as part of Amendment 162 to the PNP License and SEP Topic V-5 was closed.

The PNP PCS Leak Detection licensing basis describes the following for Containment Area Radiation - One radiation monitor, sensing from the discharge of all operating containment air coolers, is capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel (UFSAR 4.7.1.3).

(iii) monitoring condensate flow rate from air coolers.

PNP TS 3.4.15.c provides the LCO and SR associated with containment air cooler condensate level switches. An excessive drain water flow from a containment air cooler coil will be indicated in the control room by an alarm. By shutting down the cooler with the indicated high water flow, it can be determined whether the coil is leaking or if excessive condensate is being formed. A steam leak or primary coolant leak would be accompanied by an increase in the containment atmosphere humidity which would be detected by the containment humidity sensors and indicated in the control room. The absence of humidity indication would indicate the excessive water is from a leaking coil (Reference 3, Section 6.3.3.7). Each containment air cooler design includes a sump pan with a drain, a liquid level switch (LS-0817, 0865, 0870, 0868) and an overflow path. Normally very little water will be condensed from the containment atmosphere and the small amount of condensate will easily flow out through the sump pan drain. If leakage flow to the sump pan is greater than 20 gpm, the level in the sump pan will rise to the liquid level switch and trigger an alarm in the control room at +6” from the bottom of the sump pan. TS 3.4.15.c requires one of the air cooler condensate level switches to be operable. Operator action for a containment air cooler sump pan level alarm includes referencing Abnormal Operating Procedures (AOPs) and referring to TS LCO 3.4.15.

(iv) monitoring primary coolant system inventory balance as required by TS.

PNP TS SR 3.4.13.1 requires Operators verify PCS operation LEAKAGE is within limits by performance of PCS water inventory balance. As described in response to Regulatory Position 2.1, the PNP inventory balance leak rate calculation capability has evolved such that either by manual (hand calculation) or automated PPC means; total, identified and unidentified leak rate are calculated and recorded to the nearest 0.001 gpm.

In addition to the monitoring systems detailed in the TS, the plant should use other systems to detect and monitor for leakage, even if it does not have the capabilities specified in Regulatory Position 2.2. These supplemental instruments/methods may include, but are not limited to, the following:

(a) *monitoring airborne gaseous radioactivity*,

PNP TS 3.4.15.b provides the LCO and SR associated with containment atmosphere gaseous radiation monitoring. One radiation monitor, sensing from the discharge of all operating containment air coolers, is capable of detecting a 100 cm^3/min leak in 45 minutes based on 1% failed fuel (100cm^3/min is equivalent to ~0.03 gpm). RIA-1817 is required to be operable per TS 3.4.15.b and operability is verified each shift by control room operators. RE/RIA-1817 actuates a control room gaseous waste monitoring hi radiation alarm at 4130 cpm (warning) and 5130 cpm (high). Operator action is invoked by plant AOPs.

Pursuant to NUREG-0737, *Clarification of TMI Action Plan Requirements*, two high range gamma monitors have been installed in the containment building at the perimeter of the containment dome. The monitors are ion chambers with the readout range extended to 10^7 R/hour. These monitors are designed to provide a continuous readout of containment radiation levels for all conditions ranging from normal operation to hypothetical accident conditions. RE-2321/2322 actuate control room containment gamma alert and high radiation alarms at 40 R/hr (alert) and 400 R/hr (high). Operator action is directed by plant AOPs. These monitors are safety related, environmental and seismic qualified (Reference 3, Appendix 7C).

(b) monitoring the humidity of the containment,

PNP TS 3.4.15.d provides the LCO and SR associated with Containment atmosphere humidity monitoring. Four humidity transmitters (HT-1812, 1813, 1814 and 1815) are provided in the containment building to detect leakage from the PCS and the main steam lines. The relative humidity measured by these detectors is indicated in the control room (Reference 3, Section 9.8.2.1). Each of 4 humidity detectors (1 in each steam generator compartment, 1 at the refueling floor near the reactor refueling cavity and 1 near the containment dome) is capable of detecting a change of humidity of 10% which would result from approximately 150 gallons of primary water leakage (Reference FSAR 4.7.1.1). This equals a leak rate sensitivity of 2.5 gpm in one hour (Reference 10, Table 1). There are no annunciators associated with the humidity indication in the control room and no indication on the PPC.

(c) monitoring the temperature of the containment,

Four instruments are provided for monitoring the containment atmosphere temperature throughout the predicted accident range. Reactor cavity, steam generator space and containment dome temperature are indicated on the main control board up to 400°F. The system provides continuous display even though this is not a requirement. Sensors are resistance temperature detectors placed in appropriate locations in the containment. TI-1812, 1813, 1814, and 1815 provide control room panel indication with no alarm function. Containment Dome Temperature (TE-1815) is monitored hourly by Control Room Operators with a resolution accuracy of 0.01 Degree F.

(d) monitoring the pressure of the containment,

Two redundant Class 1E continuous wide-range pressure transmitters (PT-1812A, PT-1805A) fed from separate preferred ac power sources are installed and transmit signals to two recorders located in the control room with state-of-the-art accuracy and response time characteristics. The monitors have a range of 0 psia to 200 psia enveloping the range of -5 psig to at least three times the design pressure of the containment building (Reference UFSAR 7.4.6.2). PT/PI-1814 actuates a control room containment pressure off normal alarm at 0.8 +/- 0.2 psig. Operator action is defined in plant procedures. Containment Pressure (PT-1812A) is monitored hourly by Control Room Operators with a resolution accuracy of 0.01 psig using the PPC.

(e) monitoring acoustic emission, and

PNP has acoustic monitoring installed for Pressurizer Power Operated Relief Valves (PORV) and code safety valves monitored in the control room. No documentation was found stating a leak rate sensitivity. There is also installed temperature monitoring in place to detect valve seat leakage. The Attachment to this enclosure summarizes the PORV and safety valve acoustic monitoring capability.

(f) conducting video surveillance.

None

2.4. At least one of the leakage monitoring systems required by the plant TS (as described in Regulatory Position 2.3 above) should be capable of performing its function(s) following any seismic event that does not require plant shutdown.

PNP UFSAR Appendix 7C provides a listing and an evaluation of all instrumentation contributing toward meeting Regulatory Guide 1.97, *Instrumentation for Light-Water-Reactor-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and Following an Accident,* Revision 3, May 1983. Of the PCS leakage monitoring systems required by TS, only the containment sump level indications instruments LT-0382/0383 are identified in Appendix 7C. These instruments are Type B Category 2 and; therefore, comply with the Quality Assurance (QA) requirement and Environmental Qualification. Exceeding requirements, both LT-0382 and LT-0383 also have Seismic Category 1 and Electrical Class 1E classifications. Therefore, the sump level instruments meet Regulatory Position 2.4.

Of the non-TS required containment instrumentation available for PCS leak detection, the following are listed in Appendix 7C.

• Containment Pressure PT-1812A PT-1805A safety related, environmental and seismic, Preferred redundant 1E. Available on PPC.

• Containment Area Radiation RE/RIA-2321, 2322 safety related, environmental and seismic, Preferred redundant 1E. Available on PPC.

• Containment Temperature TE-1812-1815 provided for routine surveillance. Not safety related, environmental and seismic qualified, nor Preferred redundant 1E. However, containment dome temperature indicator TI-1815 does have a Seismic Category 2 classification and is available for monitoring and trending on the PPC.

2.5 The leakage monitoring systems, including those with location detection capability, should have provisions to permit calibration and testing during plant operation to ensure functionality or operability, as appropriate.

TS 3.4.15 Surveillance Requirements specify calibration and testing (channel checks) to ensure functionality or operability in accordance with the PNP Surveillance Frequency Control Program.

3. Operations-Related Positions

3.1. *The plant should periodically analyze the trend in the unidentified and identified leakage rates. When the leakage rate increases noticeably from the baseline leakage rate, the plant should evaluate the safety significance of the leak. The plant should determine the rate of increase in the leakage to verify that plant actions can be taken before the plant exceeds technical specification limits.*

Control Room Operators monitor Containment Sump Level via PPC or LIA-0359 and Component Cooling Water Surge Tank level on an hourly basis. Control Room Operators also monitor PPC Trend Group 11 hourly. PPC Trend Group 11 consists of a fixed set of real-time data trends for 60 different instruments, many of which are associated with PCS leakage detection and have been identified in the information provided for Regulatory Positions 1 and 2.

Control Room Operators monitor the following containment parameters on a once-per-shift basis.

• Containment Sump Level (LT-0382, 0383)

• Humidity (HI-1812, 1813, 1814,1815)

• Pressure (PIA-1814, 1815)

• Gas Monitor (RIA-1817)

• High Radiation (RIA-1805, 1806, 1807,1808)

Control Room Operators perform a PCS Leakage Calculation by performance of a PCS water inventory balance as required by TS SR 3.4.13.1.

Any adverse indications or trends indicative of PCS Leakage are evaluated in accordance with Admin 4.19.

3.2 The plant should establish procedures for responding to leakage. These procedures should address the following considerations and should ensure that no adverse safety consequences result from the leakage:

(i) Plant procedures should specify operator actions in response to leakage rates less than the limits set forth in the plant TS. The procedures should include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring, verifying the leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walkdown outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage.

The response to General Position 1.1 describes the principal procedures utilized by Plant Operators to ensure TS LCOs and SRs for PCS Operation LEAKAGE and PCS Leakage Detection Instrumentation are met. Additional procedures utilized by Operators to monitor for, identify and quantify PCS Operational LEAKAGE include:

• SHO-1 *Operators Shift Items* Section

• Various Alarm Response Procedures (ARP) that describe immediate and follow up operator actions to Control Room alarms

• ARP-8 Attachment 1 “Containment Sump Level Changes”

• ARP-8 Attachment 2 “Gaseous Waste Monitoring High Radiation”

• AOP-23 “Primary Coolant Leak”

(ii) Plant procedures should specify the amount of time the leakage detection and monitoring instruments ***(other than those required by TS****) may be out of service to ensure that the leakage rate is effectively monitored during all phases of plant operation (i.e., hot shutdown, hot standby, startup, transients, and power*

operation).

Leakage detection and monitoring instruments referenced by ADMIN 4.19 or AOP-23, other than those required by TS 3.4.15, that have limiting conditions of operation include:

• Power Operated Relief Valves (PORV) Position Indication – Temperature

Functionality of the PORV position indication is controlled by the PNP Operating Requirements Manual (ORM), similar to how TS required equipment is controlled by TS LCOs. These components are calibrated per TS Surveillances.

• Safety Injection Tank (SIT) Level

SIT Level Switches are directly associated with tank operability. The float-style level switches are calibrated per TS Surveillance Procedure at a refueling interval. Trending level transmitters on the tanks are verified in calibration with the level switches and are empirically checked every month during SIT sampling. Recalibration of the components can be completed upon demand.

• Containment Pressure

Containment Building pressure is checked every shift per Operations surveillance. These components are calibrated every refueling as found and as left. These components alarm in the main control room.

• Containment High Radiation Channels

Containment Area Radiation Monitors RIA-1805, RIA-1806, RIA-1807, and RIA-1808 are required by TS section 3.3.3, *Engineered Safety Features (ESF) Instrumentation*. As such, inoperability is limited and controlled. These radiation monitors are calibrated, and functionally checked each refueling outage. These have alarm capabilities in the Control Room and on the PPC.

Out of service plant equipment, including instrumentation not covered by TS LCOs, is controlled by the PNP Operations Control of Equipment Administrative Procedure. The procedure controls risk either quantitatively or qualitatively, directs Operations’ interface with the Work Week Process and assigns risk to activities. PNP utilizes an integrated risk assessment which works with the online work management process. These two processes are used in conjunction with the work management process by Operations to prioritize non-TS equipment for expedited repair, such as the PCS leakage detection system components identified in the Attachment to this enclosure. Consideration is given for timely response to important equipment issues. The work request screening process details how plant personnel will establish priority for each work order. Based on the priority assigned, the work will enter the repair process.

3.3 The plant should provide output and alarms from leakage monitoring systems in the main control room. Procedures for converting the instrument output to a leakage rate should be readily available to the operators. (Alternatively, these procedures could be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring systems should take place. The alarm should provide operators an early warning signal so that they can take corrective actions, as discussed in Regulatory Position 3.2 above.

Procedure DWO-1 is utilized to convert instrument output and other manually collected plant data to a leakage rate. DWO-1 specifies how to perform the leak rate calculation by hand calculation as well as how to utilize the PPC automatic calculation.

The PPC utilizes calibrated instrumentation to provide operators with real time PCS Leakage calculations including continuous PCS leakage monitoring, Primary to Secondary leakage calculations, and Containment Sump fill rate calculations. The control room also has Control Rod Drive Seal Leak Off Temperature monitoring and Primary Coolant Pump Controlled Bleed Off Flow and Temperature monitoring.

Inputs to the PCS leak rate calculation on the PPC include:

• Pressurizer Level – calibrated at a refueling interval. These components have Control Room alarms and as well as PPC indications.

• Pressurizer Pressure - calibrated at a refueling interval. These components have Control Room alarms and as well as PPC indications.

• Volume Control Tank Level – calibrated per work order at a refueling interval. This is monitored at a minimum of hourly by Operators and can be isolated, backfilled, and calibrated, at any time if its indication is in question.

• Primary Coolant System Average Temperature (TAVE) – calibrated by surveillance at a refueling interval. There are two independent channels with digital indication continuously available on the control panel and PPC.

Several radiation monitors assist in PCS leakage monitoring. All these radiation monitors have surveillance procedures validating their calibration as required by the Offsite Dose Calculation Manual (ODCM).

• RIA-0631, Condenser Off gas Radiation Monitor and off gas flowrate indication, is calibrated by surveillance and functionally checked quarterly, and source checked monthly.

• RIA-0707, Steam Generator Blowdown Monitor, is calibrated by surveillance and functionally checked quarterly and source checked monthly.

• RIA-2323 and RIA-2324, Main Steam Line Radiation Monitors, are calibrated by surveillance and functionally checked quarterly and source checked monthly.

• RIA-1817, Containment Atmosphere Gas Monitor, is calibrated by surveillance RR-9M, *Containment Atmosphere Gas Monitor RIA-1817 Calibration and Channel Functional Test*, and are monitored once per shift by Operations surveillance.

• RIA-0915, Component Cooling Water Radiation Monitor, is an early detection monitor for an intersystem Loss of Coolant Accident. It is not required by Licensing documents but is monitored hourly by Operations and is operationally checked by Chemistry personnel quarter per procedure. Out-of-service time for this monitor is minimal.

Containment Sump and Containment Floor indications include LT-0382, LT0383, LIT-0446A, and LIT-0446B. These components are all calibrated at a refueling interval. Indications with alarms are available on the Main Control Room Panel and PPC.

Alarms from PCS leakage monitoring components are summarized in the Attachment to this enclosure.

3.4. During maintenance and refueling outages, the plant should take actions to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action should take place to eliminate the condition resulting in the leakage.

Engineering and Operations perform walkdowns of the containment building looking for signs of PCS leakage following entry into Mode 3 during Plant shutdown. The walkdowns are performed under a work order in accordance with the outage schedule. Any new PCS leakage identified is controlled via the Corrective Action Program and outage scope control process.

4. Technical Specification Position

4.1. *Plant technical specifications should include the limiting conditions for identified, unidentified, RCPB, and intersystem leakage, and they should address the availability of various types of instruments to ensure adequate coverage during all phases of plant operation (not including cold shutdown and refueling modes of operation).*

Refer to discussion provided for General Position 1.1.

3.1.4 ***Compliance with NUREG-0800, Section 3.6.3, "Leak-Before-Break Evaluation Procedures"***

From NUREG-0800, *Standard Review Plan (SRP)*, Section 3.6.3 (Reference 17),

"The staff evaluation concludes on a plant specific and piping system specific basis that the acceptance criteria are satisfied and; therefore, that dynamic effects of pipe rupture may be eliminated from design consideration."

The staff determination is based on the following:

1. That water hammer, corrosion, creep, fatigue, erosion, environmental conditions, and indirect sources are remote causes of pipe rupture.

The NRC SE for the approved CEOG LBB Analysis for the PCS piping at PNP (Reference 2) acknowledged there is low susceptibility to cracking failure from these conditions at PNP. It states:

> "For CEOG facilities, there is no history of cracking failure in reactor coolant system (RCS) primary loop piping. The RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle)."

The full list of NRC findings in their SE for the CEN-367-A analysis (Reference 2) is provided in Section 3.1.1, above. More information on the material conditions of the PCS hot and cold leg piping, aging impacts, and how the PCS piping is maintained is included in Section 3.1.2 above. In addition, the following information is provided to close any gaps which may exist to fully address this review criteria.

<u>Screening Criteria for Degradation Mechanisms</u>

NUREG 0800, Section 3.6.3 specifies that the piping requested for the LBB application should not experience active degradation mechanisms such as erosion/corrosion (wall thinning), stress corrosion cracking (SCC), water hammer, creep and cleavage failure, brittle failure, cycle fatigue, thermal stratification and aging. These requirements are similar to the staff evaluation criteria outlined in Section 2.2 of CEN-367-A. Below is an evaluation of each of the NUREG 0800, Section 3.6.3 criteria items.

Erosion/Corrosion (Wall Thinning)

Significant parameters affecting erosion-corrosion are the chemical composition of the pressure boundary material, pH level, temperature and oxygen content of the coolant, and coolant flow linear velocity and turbulence.

Stainless steel material, in particular low carbon grades which presents better Intergranular Stress Crack Corrosion (IGSCC) resistance, is highly resistant to the erosion/corrosion mechanisms. As chromium and molybdenum content increase, wear rate decreases. The austenitic stainless steels essentially are immune to erosion/corrosion (wall thinning).

Erosion-corrosion is most likely to occur in minimum flow-recirculation lines, downstream of flow control valves and in elbows in close proximity to other fittings. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by the control rod positions; pressure is also controlled within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. Water chemistry, temperature, oxygen content are monitored and control in the primary pipes. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics are controlled in the design process. Additionally, reactor coolant system is instrumented to verify the flow and vibration characteristics of the system.

The improved flow geometries design of these lines as well as the knowledge of the flow and the characteristics of primary water mean that erosion/corrosion is not a concern for the PCS.

Stress Corrosion Cracking

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment.

In PNP, the reactor coolant piping is of rolled bond clad construction, with a base metal of ASTM A 516, Grade 70, with a cladding of 304L stainless steel with a nominal thickness of ¼-inch.

Since some residual stresses and some degree of material susceptibility exist in any stainless steel cladding, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment as well as other material in the system, and applicable ASME Code rules.

Strict pipe cleaning standards prior to operation is used to prevent the occurrence of a corrosive environment (presence of oxygen, fluorides, chlorides, sulfur forms, hydroxides, hydrogen peroxide). During plant operation, water chemistry, pH and conductivity are carefully controlled (monitored and maintained) in accordance with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation.

Contaminant concentrations are kept below the thresholds known to be conducive to SCC minimizing the likelihood of appearance of this phenomenon.

In addition, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) 236 (Reference 24) concluded that the potential for SCC in PWR primary system piping is extremely low if the water chemistry is managed per procedure.

PWSCC

The primary coolant system (PCS) hot leg (HL) and cold leg (CL) piping and the HL and CL to Reactor Pressure Vessel (RPV) welded connections are clad on the inside with austenitic stainless steel (References 6 and 23). There is no alloy 600 material in the reactor coolant pump (RCP) to loop piping attachment welds (References 6 and 22). There is no alloy 600 material in the PCS piping to SG welded connections (Reference 22).

The likelihood of PWSCC in the PCS main loop piping is eliminated due to the absence of any Alloy 600/82/182 materials of construction in the PCS loop piping and connection welds.

PWSCC at locations beyond the piping and welded connections of interest in this evaluation are managed by the Nickel Alloy Program (Reference 23).

A manual search through outage inspection reports in the PNP database yielded no indications of flaws or leakage in the PCS piping throughout the life of the plant.

Water hammer

The potential for water hammer in the PCS piping is low because they are designed and operated to preclude a voiding condition and the PCS has no valves. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the RCP characteristics are controlled in the design process. CEN-367-A evaluation concluded that the PCS piping is not prone to water hammer.

Thermal and Irradiation Aging

Research studies of thermal aging embrittlement of carbon steels have shown that the long-term thermal aging embrittlement of these steels at PWR temperatures of 572°F (300°C) is very limited (EPRI MRP-80, Reference 21). Moreover, regarding their location in the PCS, the irradiation aging phenomenon doesn't affect the PCS.

Since the ferrite volume fraction in the welds is maintained in the range required by ASME Code rules, the thermal ageing embrittlement of austenitic stainless steel welds is managed in normal PCS conditions. In general, unirradiated austenitic stainless steel base metals possess a high degree of fracture toughness and fracture is preceded with extensive plastic deformation (MRP-80, Reference 21). Hence, thermal aging embrittlement of austenitic stainless steel base metal is not a concern. Hence, thermal aging embrittlement of austenitic stainless steel base metal is not a concern.

Therefore, the PCS is not susceptible to the thermal and irradiation aging effect.

Creep and Cleavage Failure

Table 4-1 of the UFSAR lists the PCS CL and HL temperatures at 545 °F and 591°F respectively. This is well below the temperature that would cause any creep damage in SA 516 piping (which starts to appear at approximately 800°F). Cleavage type failures are not a concern for the material used at these operating temperatures.

Brittle Fracture

Brittle fracture for SA-516 Grade 70 occurs when the operating temperature is about -20°F. Brittle fracture for stainless steel material occurs when the operating temperature is about -200°F. PCS piping operating temperature is higher than 80°F and therefore, brittle fracture is not a concern for the PCS piping.

Low Cycle and High Cycle Fatigue

CEN-367-A evaluation concluded that the PCS piping system is not prone to low cycle and high cycle fatigue.

Thermal Stratification

Thermal stratification occurs when conditions permit hot and cold layers of water to exist simultaneously in a horizontal pipe. This can result in significant thermal loadings due to the high fluid temperature differentials. Changes in the stratification state result in thermal cycling which can cause fatigue damage. The thermal stratification phenomenon may be of concern in branch piping systems connected to the PCS loop. Thermal stratification is not of concern to PNP PCS piping system with leak before break because the flow pattern is continuous and the coolant is well mixed such that thermal stratification is unlikely to occur.

2. That a deterministic fracture mechanics evaluation has been completed and approved by the staff.

CEN-367-A is an LBB analysis performed in accordance with 10 CFR 50, Appendix A, GDC 4,
NUREG-1061, Volume 3. It was approved for use and acknowledged by the staff as bounding for PNP. The Reference 2 SE NRC staff statements listed in Section 3.1.1 above give the reasoning used for approval in meeting the requirements of GDC 4 and are consistent with the guidance in NUREG-0800, Section 3.6.3.

The following findings are particularly salient to demonstrating adherence to the SRP Section 3.6.3 guidance.

(1) Normal operating loads, including pressure, deadweight, and thermal expansion, were used to determine leak rate and leakage-size flaws. The flaw stability analyses performed to assess margins against pipe rupture at postulated faulted load conditions were based on normal plus SSE loads, in the stability analysis, the individual normal and seismic loads were summed to maximize the postulated flaw opening area. Leak-before-break evaluations were performed for the limiting location in the piping.

(5) In the flaw stability analyses, the staff evaluated the margin in terms of load for the leakage-size flaw under normal plus SSE loads. The results of the CEOG analyses, consistent with the results of the staff's calculations, indicated the margin exceeded 1.4 when the individual normal and seismic loads were summed. The margin is consistent with the guidelines of NUREG-1061, Volume 3 (Reference 4).

(6) Similar to item (5) above, the margin between the leakage-size flaw and the critical-size flaw was also evaluated in the flow stability analyses. The results of the CEOG analyses, consistent with the results of the staff's calculations, indicated the margin in terms of flaw size exceeded 2. The margin is consistent with the guidelines of NUREG-1061, Volume 3 (Reference 4).

The NRC letter approving CEN-367-A (Reference 2) includes the statement, “We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved.”

3. That leak detection systems are sufficiently reliable, redundant, diverse and sensitive, and that margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

The above condition is shown to be met by demonstrating consistency with RG 1.45 (Reference 18). Section 3.1.3.2 above lists all the Regulatory Positions of

RG 1.45 and explains how each is met. Note that position 2.3(ii) is the only one not explicitly met, however as is explained in Reference 12 in Section 4.15,

"Although all of the leakage-detection systems recommended by the Regulatory Guide are not present, PNP incorporates six additional diverse systems. Taking all of these systems into consideration, it is the staff's judgment that a 1-gpm leak from the RCPB to the containment will most probably be detected within 24 hours."

This alternate means to meet the guidance was therefore accepted as adequate.

The staff noted in the SE for CEN-367-A (as stated in Section 3.1.1) regarding margin to detect the through-wall flaw,

> "The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems; the margin is a factor of 10 in leakage and is consistent with the guidelines of NUREG-1061, Volume 3 (November 1984)."

Therefore, margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

3.2 <u>Evaluation Findings</u>

1) The technical justification for crediting LBB to the PNP licensing basis is found in CEN-367-A. The SE for CEN-367-A verifies its applicability to PNP.

2) Physical plant changes made to PNP and aging since the SE for CEN-367-A was approved have not invalidated the acceptability of CEN-367-A application to PNP.

    a. The only significant plant change at PNP is the replacement of the SGs. The LBB analysis based on the original conditions should remain valid, with no significant changes to the dynamic effects or material properties that would necessitate a revision of the analysis.

    b. ISI and TLAA, and Nickel Alloy and ISI programs instituted at PNP are effective and assure that the PCS piping remains in a condition free of age-related degradation mechanisms.

    c. PCS piping is not susceptible to PWSCC

3) The PCS leak detection systems at PNP are consistent with RG 1.45 guidance.

The LBB analysis does not include any time dependencies and is not being used to discontinue any activities credited for age managing components in PNPs License Renewal. Also, there are no impacts to existing calculations associated with PNP.

Therefore, application of this LBB methodology to PNP does not impact PNP License Renewal. It is; therefore, concluded that the LBB methodology is applicable to the PNP hot leg and cold leg PCS piping.

4 REGULATORY EVALUATION

4.1 <u>Applicable Regulatory Requirements/Criteria</u>

<u>Regulatory Requirements</u>

The applicable regulatory requirement for submitting the LBB evaluation to exclude the dynamic effects associated with postulated pipe ruptures from the design basis is specified in 10 CFR 50, Appendix A, GDC 4. The requirement for having a means to detect reactor coolant leakage is specified in 10CFR 50, Appendix A, GDC 30. This LAR is submitted in accordance with 10 CFR 50.90.

10 CFR 50, Appendix A GDC CRITERION 4 *Environmental and dynamic effects design bases*, states:

> *Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.*

As discussed in Section 2.3 above, the reason for this LAR submittal is to gain approval to exclude from the design basis the dynamic effects associated with postulated pipe ruptures in the PCS hot let and cold leg piping at PNP.

10 CFR 50, Appendix A GDC CRITERION 30 *Quality of reactor coolant pressure boundary,* states:

> *Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.*

As discussed in Section 3.1.3 above, in adhering to the RG 1.45 guidance, PNP had provided means of detecting, and to the extent practical, of identifying the location of the sources of reactor coolant leakage.

NUREG-1061, Volume 3 (Reference 4) provides a methodology that the NRC accepts for LBB submittals. The LBB approach described applies the fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience double-ended ruptures or their equivalent in longitudinal or diagonal splits. The NUREG also provides a step-by-step approach to performing LBB analysis. The CEOG analysis (Reference 2) followed the guidance of NUREG-1061, Volume 3, in performing the LBB analysis.

NUREG-0800, Section 3.6.3 (Reference 17) provides guidance to NRC reviewers on the specific areas to review and acceptance criteria for LBB applications. The LBB methodology and its application were reviewed in Section 3.1.4 to ensure that acceptance criteria are satisfied.

The proposed changes are consistent with the above regulatory requirements and criteria. Therefore, the proposed changes will assure safe operation by continuing to meet applicable regulations and requirements.

4.2 <u>Precedents</u>

Previously, PNP has received approval from the NRC for the use of LBB methodology for the High Thermal Performance Fuel using the submitted version of the CEN-367 CEOG evaluation (prior to approval).

1. Letter, Albert De Agazio, USNRC to Kenneth W. Berry, Consumers Power Company, Subject: “Safety Evaluation on Asymmetric LOCA Loads – MPA D-010 – Palisades,” (Reference 20).

   The above letter provides the NRCs SER for the evaluation of asymmetric LOCA loads at PNP. The majority of the component evaluations reviewed by the NRC did not credit LBB in the determination of the loads. Only the fuel assembly grid design credited LBB. The SER states:

   > “The licensee has indicated (Reference 15) that the Asymmetric Blowdown Loads issue as it affects fuel assembly grid design would be resolved by demonstrating that leak-before-break assumptions are valid for PNP. The licensee has also provided documentation (Reference 16) of the leak-before-break evaluation.“

   “Reference 15” and “Reference 16” above are References 7 and 8, respectively, in the References section of this enclosure.

The NRC has approved similar LARs or letter submittals to allow for the acceptance of LBB methodology analysis as listed in precedents below.

The following precedents were submitted by other CE plants and received approval by the NRC for which the CEOG LBB analysis was used as their basis (Reference 2):

2. Letter from George Kalman (NRC) to Jerry W. Yelverton (Entergy Operations, Inc.), “Containment Leak Detection Capabilities with

Permanent Reactor Vessel Seal Plate at Arkansas Nuclear One, Unit 2 (TAC No. M90610)," dated June 18, 1996, (ADAMS Accession Numbers ML2011E466 and ML2011E470).

3. Letter from Patrick D. Milano (NRC) to Martin L. Bowling, Jr. (Northeast Nuclear Energy Company), "Revised Evaluation of the Primary Cold Leg Piping Leak-Before-Break Analysis for the Millstone Nuclear Power Station, Unit No. 2 (TAC No. MA1070)," dated November 9, 1998, (ADAMS Accession Numbers ML20195B763 and ML20195B871).
4. NRC Safety Evaluation, "Safety Evaluation of the Office of Nuclear Reactor Regulation Related to Application of Leak-Before-Break to Reactor Coolant System Piping Southern California Edison Company San Onofre Nuclear Generating Station, Units 2 and 3 Docket Nos. 50-361 and 50-362," dated April 11, 1996, (ADAMS Accession Number ML20107B709).
5. Letter from Jan A. Norris (NRC) to Mr. J. H. Goldberg (Florida Power and Light Company), "St. Lucie Units 1 and 2 – Application of Leak-Before-Break Technology to Reactor Coolant System Piping – TAC Nos. M84560 and M84561," dated March 5, 1993, (ADAMS Accession Numbers ML20138E042 and ML20136D807).

The following approval was given for a CEOG plant depending on the Reference 2 CEOG evaluation where RG 1.45, R0 requirements remained the plant's Licensing Basis after R1 was released.

6. Letter from N. Kalyanam (NRC) to Vice President Operations (Entergy Operations, Inc.), "Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Re: Approval of Leak-Before-Break of the Pressurizer Surge Line (TAC No. ME3420)," dated February 28, 2011, (ADAMS Accession Number ML110410119).

The following approval was given for a Westinghouse plant for the purpose of removing consideration of the dynamic effects associated with the postulated rupture of the pressurizer surge line piping from the licensing basis:

Letter from John F. Stang (NRC) to Robert B. Powers (Indiana Michigan Power Company), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Review of Leak-Before-Break for the Pressurizer Surge Line Piping as Provided by 10 CFR Part 50, Appendix A, GDC 4 (TAC Nos. MA7834 and MA7835)," Dated November 8, 2000, (ADAMS Accession Number ML003767675).

4.3 No Significant Hazards Consideration

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), 50.90, *Application for amendment of license, construction permit, or early site permit*, Holtec, on behalf of Holtec PNP, LLC (Holtec PNP), hereby requests U.S. Nuclear Regulatory

Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) licensing basis to credit Leak Before Break (LBB). This LAR is consistent with the PNP LAR, dated December 14, 2023, to revise the Permanently Defueled Technical Specifications (PDTS) to reflect the resumption of power operations at PNP, which is currently under NRC review. This LAR proposes no Technical Specification (TS) changes; however, Licensing Bases changes, including the Power Operation Licensing Basis (POLB) Updated Final Safety Analysis Report (UFSAR), will be made under the 10 CFR 50.59 review process upon LAR approval.

Holtec 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(c), "Issuance of amendment," as discussed below:

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

   Response: No.

   Overall protection system performance will remain within the bounds of the previously performed accident analyses. The design of the protection systems will be unaffected. The reactor protection system and engineered safeguard systems will continue to function in a manner consistent with the plant design basis. All design, material and construction standards that were applicable prior to the request are maintained.

   For PNP, the bounding accident for pipe breaks is a Large Break Loss of Coolant Accident (LBLOCA). Since the application of the LBB analysis verifies the integrity of the Primary Coolant System (PCS) hot leg and cold leg piping and the piping integrity has been maintained since the approval of the LBB analysis, the probability of a previously evaluated accident is not increased. The consequences of a LBLOCA have been previously evaluated and found to be acceptable. The application of the LBB analysis will cause no change in the dose analysis associated with a LBLOCA, and therefore, does not affect the consequences of an accident.

   The proposed amendment will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the POLB UFSAR as restored.

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

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

   Response: No.

   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety related systems or components and does not challenge the performance or integrity of any safety related system. Further, there are no changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of power operation or change any operating parameters.

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

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

Response: No.

Margin of safety is related to the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment. The proposed amendment request does not involve a change to any of these barriers.

The proposed amendment does not involve a significant reduction in a margin of safety because the proposed amendment does not reduce the margin of safety that exists in the PNP POTS or the POLB UFSAR to be restored. The operability requirements of the TSs are consistent with the initial condition assumptions of the safety analyses.

This proposed licensing basis change uses LBB technology combined with leakage monitoring to show that it is acceptable to exclude the dynamic effects associated with postulated pipe ruptures in the PCS piping from the licensing basis. The referenced Combustion Engineering Owner's Group (CEOG) analysis (CEN-367-A, *Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems*) demonstrates that the LBB margins discussed in NUREG-1061, *Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks*, Volume 3, are satisfied.

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

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

4.4 Conclusions

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

5 ENVIRONMENTAL CONSIDERATION

This LAR requests approval for application of a leak-before-break (LBB) methodology to piping for the large bore PCS piping at PNP. This proposed change would eliminate the need to account for the dynamic effects associated with high-energy pipe rupture in the primary coolant system from the licensing and design bases of PNP consistent with 10 CFR 50, Appendix A, General Design Criteria (GDC) 4. Consistent with this LAR's No Significant Hazards Consideration in Section 4.3; i) The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, ii) The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated, and iii) The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

This LAR falls within the scope of the *Draft Environmental Assessment and Draft Finding of No Significant Impact for the Palisades Nuclear Plant Reauthorization of Power Operations Project*, Issued January 2025, (ADAMS Accession Number ML24353A157) page 1-2, *Introduction*, as "… other regulatory or licensing requests submitted to the NRC that are necessary to reauthorize power operations of Palisades…" which is part of the NRC review process for the suite of licensing actions which will allow Palisades to resume power operations.

As such, 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 REFERENCES

1. Letter, Jean A. Fleming, Holtec International, to US Nuclear Regulatory Commission (NRC) Document Control Desk, Subject: “License Amendment Request to Revise Renewed Facility Operating License and Permanently Defueled TS to Support Resumption of Power Operations,” dated December 14, 2023, ML23348A148.
2. CEN-367-A, Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems, February 1991, ML20070S390.
3. Palisades Updated Final Safety Analysis Report (UFSAR), Revision 35, ML21125A285.
4. NUREG-1061, Volume 3, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks,” November 1984.
5. Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” Revision 0, May 1973.
6. SEP-ISI-PLP-003, Revision 11, “Program Section for Palisades Inservice Inspection Master Program Fifth Interval, ASME Section XI, Division 1”.
7. Letter, Consumers Power to NRC, “Docket 50-255-License DPR-20 - Palisades Plant - Final Disposition of Asymmetric Blowdown Loads Issue as it Affects Fuel Assembly Grid Design (TAC No. M08621),” dated March 16, 1988, ML18054A284.
8. Letter CEOG-87-662, J.K. Gasper, Chairman of the CE Owners Group to James A. Norberg (NRC), Subject: “Applicability of Leak-Before-Break to the Primary Coolant System Piping of C-E Designed PWRs,” dated November 20, 1987, ML20236U814.
9. NUREG-1871, “Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant,” January 2007.
10. Letter, NRC to Consumers Power Company, “Palisades - SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection,” dated February 4, 1982, ML18046B254.
11. Letter, Juan Ayala, NRC to Paul A. Harden, Nuclear Management Company, LLC, Entergy Nuclear Operations, “Issuance of Renewed Facility Operating License No. DRP-20 for Palisades Nuclear Plant,” dated January 17, 2007, ML070100476 and ML070100463.
12. NUREG-0820, “Integrated Plant Safety Assessment Systematic Evaluation Program, Final Report”, dated October 1982, ML18047A670
13. Letter, David J. VandeWalle, Consumers Power Company to Dennis M Crutchfield, NRC; Subject: “Docket 50-255 - License DPR - 20 - Palisades Plant - SEP TOPIC III-5.A ‘Operability of Leakage Detection Systems’ –

Request for Change to the Technical Specifications," December 13, 1983, ML18051A752.

14. Letter, David J. VandeWalle, Consumers Power Company to Director, Nuclear Reactor Regulation (NRR); Subject: "Docket 50-255 - License DPR - 20 - Palisades Plant - Proposed Technical Specification Change Request - Leakage Detection System," May 23, 1985, ML18051B398.
15. Letter, Gerald B. Slade, Consumers Power to NRC Document Control Desk, Subject: "Docket 50-255 - License DPR-20 - Palisades Plant - Technical Specification Change Request - Instrumentation and Controls," ML18057B377, ML18057B379
16. Letter, Marsha K. Gamberoni, NRR to Robert A. French, Consumers Power Company, Subject: "Palisades Plant - Issuance of Amendment Re: Instrumentation Operability Requirements (TAC No. M82124)," dated October 26, 1994, ML020840096.
17. NUREG-0800, Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1, March 2007.
18. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 1, May 2008.
19. Westinghouse PWR Owners Group Letter OG-07-286, "Recommendations for Implementation of Guidelines Defined in 'Standard Process and Methods for Calculating RCS Leak Rate for Pressurized Water Reactors' (WCAP-16423-NP, Rev 0) and 'Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors' (WCAP-16465-NP, Rev 0) (PA-OSC-0189 and PA-OSC-0218) with respect to NEI-03-08," dated October 19, 2006; ML070310081, ML070310084, ML070310082.
20. Letter, Albert De Agazio, NRC to Kenneth W. Berry, Consumers Power Company, Subject: "Safety Evaluation on Asymmetric LOCA Loads - MPA D-010 - Palisades (TAC No. M08621)," October 27, 1989, ML18054B068 with Attached Safety Evaluation ML19325E446.
21. EPRI, "Materials Reliability Program: A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80) 1003523 Final Report", May 2003.
22. EPRI, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139)," August 2005, ML060170529.
23. SEP-A600-PLP-001, Revision 2, "Nickel-Alloy Program – Entergy Nuclear Engineering Programs".
24. EPRI, "Materials Reliability Program: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors (MRP-236, Rev. 1)."

25. Holtec Decommissioning International, LLC letter to U.S. Nuclear Regulatory Commission, “Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments,” dated December 6, 2023 (ADAMS Accession No. ML23340A161)

7 ATTACHMENT

RCPB Leakage Detection Systems RG 1.45 Requirements.

Attachment to Enclosure

PNP 2025-002

RCPB Leakage Detection Systems
RG 1.45 Requirements

4 Pages Follow

Table 1 RCPB Leakage Detection Systems RG 1.45 Requirements – RCPB to Containment

System Component	Instrument / Point ID	Leak Rate Sensitivity	Safety Related	EQ	Seismic	1E	CR Panel	CR Alarm	PPC	PPC Resolution	Update/Scan Frequency	Monitoring Frequency	Online Test	TS or Procedure Reference
Containment Sump Level	LT-0382 LT-0383	0.01 gpm in 1 hour Note 4	Y	Y	1	Y	LPIR-0382/ 0383	Yes	Y	0.01 ft = 29.3 gal	1 sec	Hourly TG11/2		TS 3.4.15.a
Sump Pump Actuations Monitoring	None	Water is drained by gravity through two normally closed remotely operated Isolation valves. Operator action, per SOP-17B, is required to drain sump.												
Airborne Particulate Monitoring	None													
Containment Gaseous Activity Monitor	RE/RIA-1817	0.03 gpm 1% failed fuel in 45 min	QP	N	N	N	C-11 Rear	Yes EK-1364	Y	0.01 cpm	1 sec	Shiftly Hourly TG11/6	Yes	TS 3.4.15.b SHO-1 ARP-8
Air Cooler Pan Level Switch	LS-0817/0865/ 0870/0868							Yes EK-1343/ 1344/ 1345/ 1346					No	TS 3.4.15.c ARP-8 RI-117
Containment Pressure WR	PT-1812A/1805A		Y	Y	1	Y	C-08	Yes	Y	0.01 psig	1 sec	Hourly TG11/6	No	RI-6A/B
Containment Pressure NR	PT-1814/1815		Y	N	1	Y	C-13 PIA- 1814/18 15	Yes EK-1362	N			Shiftly	No	SHO-1 ARP-8 RI-6A/B
Containment Humidity	HT/HI-1812/1813/ 1814/1815	2.5 gpm in 1 hour	QP	N	2	N	Yes C-13	N	N			Shiftly	No	TS 3.4.15.d SHO-1 RI-25
Containment Temperature	TE/TI-1812/1813/ 1814/1815		QP	N	2 Note 5	Y	Yes C-13	N	1815 Dome	0.01 F	1 sec	Hourly TG11/6	No	
PORV & Safety Accoustic Emissions	FE/FM/FI- 1039/1040/1041 FE/FM/FI- 1042B/1043B	Unknown	Y	Y	1	Y	C-11A	Yes EK-1373				Weekly	No	DWO-1 ARP-8
Moisture Sensitive Tape	None													
Rx Vessel Flange Pressure	PS-0101 (Inner) PI-0101 local	130cu in = 0.56 gal	N Y	N N	N B	N N	N	Yes EK-0767 1500 psig	N				No	ARP-4
Rx Vessel Flange Level	LS-0160 (outer seal)	35 cu in = 0.15 gal	QP	N	N	N		Yes EK-0768 120" H2O						ARP-4
Safety & PORV Seat Leakage via Temperature Monitors	TE/TIA-0106 TE/TIA-0107 TE/TIA-0108 TE/TIA-0109 TIA's	Small, Value Unknown 180F	N QP	N N	N 2	N N	C-12	Yes EK-0743/ 0744/ 0745/ 0746					No	ARP-4

Table 2 RCPB Leakage Detection Systems RG 1.45 Requirements – Intersystem Leakage														
System Component	**Instrument / Point ID**	**Leak Rate Sensitivity**	**Safety Related**	**EQ**	**Seismic**	**1E**	**CR Panel**	**CR Alarm**	**PPC**	**PPC Resolution**	**Update/Scan Frequency**	**Monitoring Frequency**	**Online Test**	**TS or Procedure Reference**
Condenser Off-Gas Radiation Monitor	RE/RIA-0631	< 5 gpm	QP	N	N	Y	Yes C-13	EK-1364	Y	0.01 cpm	1 sec	Hourly TG11/3	Y	DWO-1 ARP-8
S/G Blowdown Tank Vent Monitor	RE/RIA-2320	< 5 gpm	N	N	N	N	Yes C-13	EK-1364	N				Y	ARP-8 RR-9H QR-22 MR-14
CCW Pump Radiation Monitor	RE/RIA-0915		N	N	N	N	Yes C-13	EK-1365	Y				Y	ARP-8
CCW Surge Tank Level	LT/LIA-0917	Unknown	N	N	N Note 9	N	Yes C-08	EK-1172				Hourly CR Data Sheet	Y	ARP-7
CRDMs	Seal Leak-off Line Thermocouple	<<1 gpm	N	N	N	N	Yes C-12 Rear	EK-0954					N	ARP-5

Table 3 RCPB Leakage Detection Systems RG 1.45 Requirements – Inventory Balance														
System Component	**Instrument / Point ID**	**Leak Rate Sensitivity**	**Safety Related**	**EQ**	**Seismic**	**1E**	**CR Panel**	**CR Alarm**	**PPC**	**PPC Resolution**	**Update/Scan Frequency**	**Monitoring Frequency**	**Online Test**	**TS or Procedure Reference**
Inventory Mass Balance	Hand Calc	0.001 gpm Note 10	Y	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Min 72 hrs	72 hrs	Y	TS 3.4.13 DWO-1

Table 4 RCPB Leakage Detection Systems RG 1.45 Requirements – Additional Information

System Component	Instrument / Point ID	Leak Rate Sensitivity	Safety Related	EQ	Seismic	1E	CR Panel	CR Alarm	PPC	PPC Resolution	Update/Scan Frequency	Monitoring Frequency	Online Test	TS or Procedure Reference
RCPB to Containment														
Containment Sump Level	LIA-0359	1% = 293 gal	QP	N	2	N	C13	EK-1350 2930 gallons				Manual Backup to PPC LT-0382		ARP-8
Containment Gamma Monitors	RE/RIA-2321/2322		Y	Y	1	Y	C-11A Rear	EK-0213						TS 3.3.7 ARP-33 RI-86G-1/-2 MI-6
Quench Tank Level	LT/LIA-0116		QP	N	2	Y	C-12	EK-0733	Y		1 sec			ARP-4
Quench Tank Pressure	PT/PIA-0116		QP	N	2	Y	C-12	EK-0732	Y		1 sec			ARP-4
Quench Tank Temperature	TE/TIA-0116		QP	N	N	Y	C-12	EK-0731						ARP-4
Charging Flow	FT/FIA-0212	0.01 gpm	QP	N	2	Y		EK-0735 Low Only	Y	0.01 gpm	1 sec	Hourly TG11/2		AOP-23
Letdown Flow	FT-0202/FIC-0202		QP	N	2	Y		EK-0705 High Only	Y	0.01 gpm	1 sec	Hourly TG11/2		AOP-23
Rx Head Vent Pressure	PT/PIA-1066		Y	N	1	N	Back of C-11A							RO-112
Intersystem Leakage														
Main Steam Line Radiation Monitor	RE/RIA-2323 RE/RIA-2324		QP	N	B	Y	C-11A	EK-0217	Y	0.01 cpm	1 sec	Hourly TG11/3		ARP-33
S/G Blowdown Radiation Monitor	RE/RIA-0707		N	N	N	N	C-13	EK-1365	Y	0.01 cpm	1 sec	Hourly TG11/3		ARP-8
CCW Surge Tank Level	LT/LIA-0920		QP	N	B	Y	C-08	EK-1172						ARP-7
Safety Injection Tank Level	LT/LIA-0365/0368/ 0372/0374		QP	N	2	Y	C-13	EK-1313/ 1319/ 1325/ 1331	Y	0.01%	1 sec	Shiftly Hourly TG11/10		TS 3.4.14 SHO-1 ARP-8
Safety Injection Tank Pressure	PT/PIA-0363/0367/ 0371/0369		QP	N	2	Y	C-13	EK-1316/ 1322/ 1328/ 1334						ARP-8
Inventory Balance														
Inventory Mass Balance	Automated Calc Or by Hand	0.001 gpm	Note 10	N/A	N/A	N/A	N/A	N/A	Y	0.001 gpm	Continuous On demand	N/A	Y	TS 3.4.13 DWO-1
15 Minute Avg Unidentified Leak Rate	LR_CN_15M_UNID_ AVG	0.01 gpm	Note 10	N/A	N/A	N/A	N/A	N/A	Y	0.01 gpm	1 sec	Hourly TG11/6 Note 5	Y	Admin 4.00

Notes for Tables 1 through 4:

1 - Table 1, 2 & 3 format and system components based on information credited in the 1982 NRC evaluation against RG 1.45 Revision 0, NRC Letter "Palisades SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection" dated 2/4/1982 (Reference 4). Updated information is provided in Tables 1, 2 & 3. Table 4 contains all new information.

2 - Safety Related, Environmental qualification (EQ), Seismic qualification & Electrical (1E) qualification data taken from Asset Suite unless otherwise specified. Seismic Asset Suite Classifications (per procedure FP-E-TRC-02):

> 1=Seismic Category 1 - required for safe shut down and the equipment must be seismically qualified to function post-seismic event as defined by RG 1.29
>
> 2=Seismic Category 2 - equipment is not required to function post-seismic event, but is required to maintain structural integrity post-seismic event
>
> B = Seismic Design Augmented 2 over 1 - Category 2/1 equipment is required to be seismically mounted to ensure it does not adversely impact nearby safety-related components
>
> QP=augmented quality program - extra criteria have been imposed to provide a certain level of quality above standard commercial. The most common application of augmented quality is for the fire protection system. Augmented quality for fire protection equipment means that the equipment needs to be UL/FM rated (or other applicable code). Doesn't apply to seismic aspects of equipment.

3 - Per Admin 4.00 Section 2.0 Control Room Panel walkdowns are performed hourly and PPC Trend Group (TG) 11 should be monitored frequently. There are a total of 10 PPC Trend pages in Trend Group (TG) 11. In this Table, TG11/6 means PPC Trend Group 11, Page 6.

4 - Reference 4 stated sump level monitoring sensitivity was 25 gpm. The PPC trend resolution for sump rate of rise is 0.01 gpm with an update frequency of 1 second. PCS leak rates of </= 1 gpm are readily identifiable.

5 - UFSAR Appendix 7C says TE-1815 N/A for seismic. M-319 Sh 2 Rev 3 says TE-1815 is rated to 0.217G/0.407G horizontal and 0.067G /0.113G vertical (allow/yield). Asset Suite indicates TE-1815 is seismic category 2.

8 - LIA-0359 alarms at 12 inches or 10% from bottom of sump (ARP-8). 1% = 293 gallons (ARP-8 Attachment 1). Therefore, alarm comes in at 2930 gallons addition to containment sump.

9 - Reference 4 stated LIA-0917 is rated for 0.487 g horizontal.

10 - Reference 4 stated Leak Rate Inventory Balance sensitivity was <1 gpm. Per DWO-1 leak rates are calculated and recorded to the closest 0.001 gpm. PPC calculations for inventory mass balance and unidentified leak rate implementations are covered by the PNP software quality assurance program. The PPC inventory mass balance calculation is not safety related but is augmented quality and considered a Critical Digital Asset.