UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 26, 2025

Site Vice President
Palisades Energy, LLC
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT NO. 280 RE: LEAK-BEFORE-BREAK METHODOLOGY FOR PRIMARY COOLANT SYSTEM HOT AND COLD LEG PIPING (EPID L-2025-LLA-0027)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the license in response to your application dated February 5, 2025, as supplemented by letters dated February 27, 2025, and July 10, 2025.

The amendment revises the licensing basis to include Leak-Before-Beak methodology for primary coolant system hot and cold leg piping at PNP.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Registe*r notice.

Sincerely,

/RA/

Justin C. Poole, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 280 to DPR-20
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280
Renewed License No. DPR-20

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment filed by Holtec Decommissioning International, LLC[1], on behalf of Holtec Palisades, LLC, February 5, 2025, as supplemented by letters dated February 27, 2025, and July 10, 2025, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

1 By letter dated July 24, 2025, the NRC issued Amendment No. 275, reflecting Palisades Energy, LLC, as the licensed operator (the licensee) for Palisades Nuclear Plant.

2. Accordingly, by Amendment No. 280, the license is amended to authorize the Leak-Before-Break methodology for primary coolant system hot and cold leg loop piping, as set forth in the Holtec Decommissioning International, LLC, application dated February 5, 2025, as supplemented by letters dated February 27, 2025, and July 10, 2025, and evaluated in the NRC staff's safety evaluation dated August 26, 2025. Additionally, the licensee shall update the UFSAR as described in the licensee's application and shall submit the revised description with the next update of the UFSAR.

3. This license amendment is effective upon the licensee's submittal of a request to rescind the 10 CFR 50.82(a)(1) certifications and shall be implemented within 30 days from the amendment effective date.

FOR THE NUCLEAR REGULATORY COMMISSION

ILKA BERRIOS

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Date: 2025.08.26 11:55:16 -04'00'

Ilka Berrios, Acting Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: August 26, 2025

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 280 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated February 5, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25035A216), as supplemented by letters dated February 27, 2025, and July 10, 2025 (ML25058A265 and ML25191A022, respectively), Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Palisades, LLC,[1] (collectively, Holtec), submitted a license amendment request (LAR) to make certain changes to the Renewed Facility Operating License (RFOL) DPR-20 for Palisades Nuclear Plant (Palisades, PNP).

The proposed amendment would revise the licensing basis to include Leak-Before-Break (LBB) methodology for primary coolant system (PCS) hot and cold leg piping at PNP. The basis for this request is the approved Combustion Engineering Owners Group (CEOG) evaluation CEN-367-A, "Leak-Before Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems," 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 proposed license amendment includes enhancements to the leakage detection systems including implementation of a PCS leak rate monitoring program as described in Revision 1 to Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 2008. The subject LAR requests approval for application of a LBB methodology to piping for the large bore PCS piping at PNP. The 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 title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 4.

1 On July 24, 2025, the NRC issued an order approving and conforming amendment reflecting the transfer of operating authority from HDI to Palisades Energy, LLC (Package ML25167A245). Holtec Palisades, LLC, remains the licensed owner of PNP.

The supplemental letter dated July 10, 2025, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 15, 2025 (90 FR 15730).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance on which the NRC based its acceptance and evaluation of this LAR are contained in the following subsections.

2.1 Regulatory Requirements and Guidance

2.1.1 Regulatory Requirements

As required by 10 CFR 50.36(c)(2)(i), the technical specifications (TS) will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TS until the condition can be met.

10 CFR 50, Appendix A General Design Criteria (GDC) 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.

GDC 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.

2.1.2 Regulatory Guidance

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition" (SRP), section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1 (ML063600396), provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for the application of LBB. SRP section 3.6.3 states the following:

> Leakage detection systems are evaluated to determine whether they are sufficiently reliable, redundant, and sensitive so that a margin on the detection of unidentified leakage exists for through-wall flaws to support the deterministic fracture mechanics evaluation. The specifications for plant-specific leakage detection systems inside the containment should be equivalent to those in RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" [Revision 0 (ML003740113)].

In Regulatory Guide (RG) 1.45, Revision 1 (ML073200271), the NRC staff described acceptable methods of implementing this requirement regarding the selection of leakage detection systems for the reactor coolant boundary. RG 1.45, Revision 1, regulatory position (RP) 2.3 states that plant TS 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 into TS include, but are not limited to, the following:

(i) monitoring sump level or flow,
(ii) monitoring airborne particulate radioactivity, and
(iii) monitoring condensate flow rate from air coolers.

RG 1.45, Revision 1, RP 2.2 states that 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).

2.2 Current Technical Specifications Requirements

The requirements related to the content of the TS are contained in 10 CFR 50.36, which require that the TS include LCOs. The criteria defined by 10 CFR 50.36(c)(2)(ii) relevant to determining whether the capabilities related to the reactor coolant pressure boundary leakage detection should be included in the TS LCOs, are as follows:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

In section 1 of the enclosure to its LAR, the licensee states the following:

> This LAR is consistent with the LAR, dated December 14, 2023 [(ML23348A148)], 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.

Since the submittal of this LAR, on July 24, 2025, the NRC staff issued an exemption request, a license transfer application, and four license amendment requests, including the December 14, 2023, application which restores the operational TS (ML25157A127) to support the reauthorization of power operations at PNP. The license amendments and exemption are conditioned such that the plant cannot return to an operational status earlier than August 25, 2025, which is Holtec's planned date to transition to the power operations licensing basis, as stated in its July 1, 2025, letter (ML25182A066).

Palisades' TS require periodic verification that the reactor coolant system (RCS) leakage is within limits and that the leakage detection instrumentation is operable. Specifically, TS Surveillance Requirement (SR) 3.4.13.1 requires operators to verify that the RCS operational LEAKAGE is within the limits specified in TS 3.4.13 by performance of RCS water inventory balance, and SR 3.4.13.2 requires operators to verify that the primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator (SG), each at the frequencies specified in TS 5.5.17, "Surveillance Frequency Control Program."

Palisades TS LCO 3.4.15 requires three of the following PCS leakage detection instrumentation channels to be OPERABLE:

a. One containment sump level indicating channel;
b. One containment atmosphere gaseous activity monitoring channel;
c. One containment air cooler condensate level switch channel; or
d. One containment atmosphere humidity monitoring channel.

The licensee has not proposed changes to TS with this LAR.

3.0 TECHNICAL EVALUATION

3.1 Screening Based on Applicable Degradation Mechanisms

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. The licensee stated that these requirements are similar to the staff evaluation criteria outlined in Section 2.2 of CEN-367-A.

3.1.1 Stress Corrosion Cracking

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

The licensee stated that at PNP, the reactor coolant piping is of rolled bond clad construction, with a base metal of ASTM A 516, Grade 7, with cladding of 304L stainless steel with a nominal thickness of ¼-inch.

The licensee stated that strict pipe cleaning standards prior to operation are 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.

In addition, the licensee stated that the contaminant concentrations are kept below the thresholds known to be conducive to SCC minimizing the likelihood of appearance of this phenomenon. Therefore, the likelihood of SCC is minimized during plant operation.

Therefore, the NRC concludes that stress corrosion cracking is not an active degradation mechanism for the subject piping at PNP because the type of material used in the piping along with the cladding is not prone to corrosion, and the licensee has controls in place to limit a corrosive environment such as water chemistry, cleaning of the pipe material to control fluorides, chlorides, sulfur forms, hydroxides, and hydrogen peroxide.

3.1.2 Primary Water Stress Corrosion Cracking

The licensee stated that the PCS hot leg and cold leg piping and the hot leg and cold leg-to-reactor pressure vessel welded connections are clad on the inside with austenitic stainless steel. There is no alloy 600 material in the reactor coolant pump (RCP)-to-loop piping attachment welds. The licensee also stated that there is no alloy 600 material in the PCS piping-to-SG welded connections.

The licensee concluded that the likelihood of Primary water stress corrosion cracking (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.

The licensee stated that the PWSCC at locations beyond the piping and welded connections of interest in this evaluation are managed by the Nickel Alloy Program. 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. Inservice Inspections (ISI): The licensee stated that 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 [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code (BPV Code) and applicable Code Cases, specifically ASME Code, 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: The licensee stated that 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.

In addition, the licensee performed a manual search through the outage inspection reports in the PNP database and found no indications of flaws or leakage in the PCS piping were reported throughout the life of the plant.

Accordingly, the NRC staff finds that the primary water stress corrosion cracking is not an active degradation mechanism because the PNP PCS piping under LBB consideration is not prone to corrosion due to the lack of Alloy 600/82/182 materials in PCS piping. Therefore, the staff concludes that the material degradation at PNP is managed by the licensee performing inservice inspections and repair/replacement activities as needed to eliminate the risks and concerns related to age-related material degradation.

3.1.3 CEN-367-A CEOG LBB Evaluation

The licensee stated that CEN-367-A is an LBB analysis performed in accordance with 10 CFR Part 50, Appendix A, GDC 4 and NUREG-1061, "Evaluation of Potential for Pipe Breaks, Volume 3." The analysis performed an evaluation of the CEOG plants by evaluating LBB in the hot leg and cold leg PCS piping.

The licensee stated that the CEN-367-A evaluation was approved for use at PNP on October 30, 1990. The evaluation demonstrated that the PNP primary piping system is enveloped for piping loads within their grouping and the conclusions of the generic LBB applications are applicable.

The NRC staff's previous evaluation of CEN-367-A and GDC 4 compliance in a safety evaluation dated ~~-~~February 1991 (ML20070S390)~~,~~ included the following:

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 Safe Shutdown Earthquake (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 locations in the piping.

2. For CEOG facilities, there is no history of cracking failure in 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 Argonne National Laboratory. The fracture toughness for ferritic steel was estimated based on data from NRC sponsored research at 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 all CEOG plants have RCS pressure boundary leak detection systems which are consistent with the guidelines of RG 1.45 such that a leakage of 1 gallon per minute (gpm) in one hour can be detected through a postulated flaw and used a constant factor of 250 gpm/in of flaw opening. The NRC staff did not agree with this approach. The NRC staff did calculations using a PICEP computer code and found that the CEOG 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 leakage 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 guidance will be addressed on a plant specific basis when licensees submit requests for LBB applications (See Section 3.1.13 of this SE).

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

6. The staff confirmed that the margin between the leakage-size flaw and the critical size flaw exceeded 2 which is consistent with the CEOG analyses and the guidelines of NUREG-1061, Volume 3.

The NRC staff's previous safety evaluation of GDC 4 compliance concluded the following:

> The staff has reviewed the information submitted by the CEOG and has performed independent flaw stability computations. On the basis of its review, the staff concludes that the subject CEOG primary loop piping complies with the revised GDC-4 according to the criteria in NUREG-1061, Volume 3. 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. 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.

Accordingly, the NRC Staff reviewed its prior evaluation of CEN-367-A and GDC 4 compliance as described above and concludes that it remains applicable for PNP.

3.1.4 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 loading due to the high fluid temperature differentials. Changes in the stratification state can result in thermal cycling which can cause fatigue damage. The licensee stated that thermal stratification is not a 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.

Therefore, the NRC staff concludes that thermal stratification is not an active degradation mechanism for the subject piping because hot and cold layers of water do not exist due to a continuous flow pattern and the coolants are well mixed to eliminate temperature differentials at PNP.

3.1.5 Creep and Cleavage Failure

Cleavage failure refers to brittle fracture of a material whereupon the material indicates a loss of ductility and toughness. Creep failure is a deformation or fracture of a material at a sustained stress at an elevated temperature over an extended period.

The licensee stated that the PCS Cold Leg (CL) and Hot Leg (HL) temperatures are in a range of 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). The licensee also stated that cleavage failures are not a concern for the material used at these operating temperatures.

Therefore, the NRC staff concludes that creep and cleavage failure are not active degradation mechanisms for the subject piping because the temperatures found at PNP are well below the temperature of 800°F where creep and cleavage failure become a concern.

3.1.6 Brittle Fracture

Brittle fracture is a type of material failure characterized by the sudden separation of a material into two or more parts with little or no plastic deformation.

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

Therefore, the NRC staff concludes that brittle fracture is not an active degradation mechanism for the subject piping because the operating temperatures of the PCS are well above the temperatures for SA-516 Grade 70 and stainless-steel materials where brittle fracture becomes a concern at PNP.

3.1.7 Water Hammer

Water hammer is a pressure shockwave that occurs when a fluid in a pipe is forced to change direction or stop abruptly. The impact can range from mild vibrations to disruptive pressure shocks that can damage pipes and equipment.

The licensee stated that 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. Additionally, the CEN-367-A evaluation concluded that the PCS is not prone to water hammer.

Therefore. the NRC staff concludes that water hammer is not an active degradation mechanism for the subject piping because, as stated in the NRC staff’s evaluation of CEN-367-A, the RCS primary loop at PNP has an operating history which demonstrates its inherent stability, including 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).

3.1.8 Thermal and Irradiation Aging

Thermal aging is the degradation of materials due to prolonged exposure to heat, while irradiation aging is the degradation caused by exposure to radiation. Both processes can affect the mechanical properties, microstructure, and chemical composition of materials.

The licensee stated that studies of thermal aging embrittlement of carbon steels at pressurized water reactor temperatures of 572°F is very limited. Additionally, their locations in the PCS, irradiation aging phenomenon doesn't affect the PCS.

The licensee stated that the ferrite volume fraction in the welds is in the range required by the ASME Code. The thermal aging embrittlement of austenitic stainless-steel welds is managed in normal PCS conditions and the ferrite volume fraction in the welds is maintained in the range required by ASME Section IX. Unirradiated austenitic stainless-steel base metals possess a high degree of fracture toughness and fracture is preceded with extensive plastic deformation. Therefore, the thermal aging embrittlement of austenitic stainless steel base metals is not a concern.

Therefore, the NRC staff concludes that the PCS primary piping is not susceptible to the thermal and irradiation effects because of the material properties and location of the carbon steel and austenitic stainless-steel at PNP.

3.1.9 Erosion/Corrosion (Wall Thinning)

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.

Low carbon grade stainless steel material present better Intergranular Stress Corrosion Cracking (IGSCC) resistance and is highly resistant to the erosion/corrosion mechanisms. As chromium and molybdenum content increase, wear rate decreases. The austenitic stainless steels are essentially immune to erosion/corrosion (wall thinning).

The licensee stated that 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. Temperatures during normal operation are maintained within a narrow range by the control rod positions. Pressures are also controlled within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. The licensee stated that water chemistry, temperature, oxygen content are monitored and controlled in the primary pipes.

The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters (system resistance and the reactor coolant pump characteristics) are controlled in the design process. The licensee stated that the RCS is instrumented to verify the flow and vibration characteristics of the system.

Therefore, the NRC staff concludes that erosion/corrosion (wall thinning) is not an active degradation mechanism for the subject piping because of the chemical composition of the piping and the licensees' programs in place to maintain chemistry and to monitor for linear flow in the system at PNP.

3.1.10 Inservice Inspection Program

The licensee stated that the ASME Code, Section XI ISI Program at PNP is designed to facilitate inspections and assessments to identify the 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 takes 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 licensee stated that the ISI Program is in the fifth 10-year ISI period, which began in December 2015. The ASME Code, 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. The program involves a variety of inspection techniques and covers numerous components essential to the plant's safety.

The licensee performed a manual search through outage inspection reports in the PNP database, and the results found no indications of flaws or leakage in the PCS piping throughout the life of the plant.

Therefore, the NRC staff concludes that the ISI program is acceptable for identifying degradation because the ISI program at PNP provides monitoring to identify degradation early to take necessary corrective actions. Additionally, a staff review of the database found no indications of flaws or leakage in the PCS piping for the life of PNP.

3.1.11 Fatigue

The licensee stated that the NRC extended their license in 2007 for an additional 20 years and Time-Limiting Aging Analysis (TLAA) was implemented at PNP. These programs and activities are credited for managing the effects of aging during the period of extended operation (PEO). These materials management program assure that the CEOG LBB analysis remains applicable to PNP.

As addressed in the updated final safety report (UFSAR), item g, Section 1.9.2.2, "The ASME III Class A Primary Coolant Piping Fatigue Analysis," the 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 for the period of extended operation. The licensee stated that they utilized a conservative approach to estimate stress ranges and fatigue cumulative usage factors (CUF). The HL CUF is 0.07551 and the CL CUF is 0.7531 which is below the acceptable CUF of 1.0. Additionally, the licensee also implemented a Fatigue Monitoring Program which is a proactive measure to ensure safe continued operation by addressing the risks associated with metal fatigue by managing and monitoring metal fatigue in components of the PCS pressure boundary during the PEO.

Therefore, the NRC staff concludes that fatigue is effectively managed to ensure that the CEOG LBB analysis remains applicable to PNP because the licensee utilized conservative stress ranges for CUFs which were below the acceptable CUF of 1.0, and the licensee implemented a Fatigue Monitoring Program as a proactive measure to ensure continued safe operation.

3.1.12 Steam Generator Replacement

The licensee stated that the original SGs were replaced in 1990. The replacement SGs were fabricated to the requirements of the 1977 Edition of the ASME Code. The licensee stated that the replacement SGs are consistent with the PCS. The welding of the PCS elbows was performed using the gas tungsten arc welding (GTAW) process and meets the requirements of ASME Code, Section IX; *Welding, Brazing, and Fusing Qualifications*, using the appropriate filler material. The GTAW weld process produces better fracture properties than the submerged arc welding (SAW) process. The licensee stated that 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 NRC staff concludes that the information provided above is acceptable because the SGs were fabricated with a more recent version of the ASME Code, and the materials and welding processes used are compatible with the original CEN-367-A analysis. The staff also notes that 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, the material properties remain acceptable. Additionally, the replacement steam generators do not impact the acceptability of CEN-367-A for the LBB analysis at PNP.

The licensee also stated that a comparison of the original and replacement SGs identified that the effects on piping loads, SG support loads, and the SG dynamic characteristics, are not significant. The licensee stated that the original SG bottom support skirt assembly and sliding base support were reused for the replacement SGs without modification. The licensee performed updated stress analyses for the main feedwater, auxiliary feedwater, and main steam piping and confirmed that the changes do not invalidate the original LBB analysis and the design limits for the replacement SGs are at least as conservative as for those of the original SGs.

Therefore, the NRC staff concludes that the information provided above is acceptable because the CEN-367-A evaluation remains applicable for PNP, and that the replacement SGs do not require any changes to the LBB analysis.

3.1.13 Compliance with RG 1.45

Section 3.1.3 of the LAR contains the licensee's evaluation of compliance with RG 1.45 at PNP. The licensee states that it was a condition of application of the CEOG LBB analysis in CEN-367-A 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.

The licensee reviewed PNP records and the NRC's online ADAMS to find that the earliest identified PNP references to RG 1.45, Revision 0, were associated with the PNP Integrated Plant Safety Assessment System Evaluation Program (SEP) which resulted in NUREG-0820, "Integrated Plant Safety Assessment Systematic Evaluation Program, Final Report" (October 1982, ML18047A670). In NUREG-0820, NRC stated that PNP would require a TS change to meet the RPs, 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 licensee states in the LAR that the TS Change Request, dated November 15, 1991 (ML18057B377 and ML18057B379), provided the PCS leakage detection operability and SRs needed to close SEP Topic V-5 and was approved on October 26, 1994 (ML020840096), as PNP License Amendment No. 162. The licensee states 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 No. 162 to the PNP license closed SEP Topic V-5, resulting in PNP becoming consistent (or compliant) with RG 1.45, Revision 0.

Section 3.1.3.2 of the LAR contains the licensee's evaluation on compliance with RG 1.45, Revision 1, RPs at PNP. The LCO for Palisades TS 3.4.13, "PCS Operational LEAKAGE," requires PCS operational LEAKAGE for unidentified LEAKAGE be limited to 1 gpm. RP 2.2 states that "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 NRC staff reviewed Palisades TS 3.4.15, which requires three of four PCS leakage detection channels to be operable. While the containment sump level indication was justified, the other three instruments lacked sufficient justification for detecting a 1 gpm leakage rate within 1 hour. The NRC requested additional information, and the licensee responded (ML25191A022) by identifying two independent methods meeting the requirements.

In its response to the RAI (ML25191A022), the licensee states that containment sump level (TS 3.4.15a) and PCS water inventory balance (TS SR 3.4.13.1) as discussed in the LAR under RP 2.3, provide two independent and diverse instruments and/or methods that have the detection and monitoring capabilities detailed in RP 2.2. The licensee also states that its interpretation of how many leakage detection systems should be in TS to meet RPs 2.2 and 2.3 is supported by discussion in the safety evaluation approving precedent Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, revisions to TSs to define a new time limit for restoring inoperable RCS leakage detection instrumentation to operable status (ML112082543).

The NRC staff found the response consistent with RG 1.45, Revision 1, and the staff's similar evaluation for the Calvert Cliffs Nuclear Power Plant, concluding that the PCS leakage detection system is acceptable. Accordingly, the RAI was closed. Based on its review of the information provided in the LAR, the supplement, and the NRC staff's previous review and approval of PCS leakage detection system as a part of PNP License Amendment No. 162, the staff concludes that the PCS leakage detection system conforms to the guidelines provided by RG 1.45, Revision 1, and therefore, is acceptable.

3.2 Technical Conclusion

On the basis of its review of the LAR, as supplemented, the NRC staff finds that, for the subject PCS loop piping (42-inch inside diameter (ID) HL pipe from the reactor vessel outlet to the SG inlet and lengths of 30-inch ID cold leg pipe between the SG outlet and the PCP suction nozzle and between the PCP discharge and the reactor vessel inlets, the licensee has demonstrated that the criteria of SRP Section 3.6.3 have been satisfied in the evaluation of the potential degradation mechanisms such as erosion/corrosion (wall thinning), SCC, water hammer, creep and cleavage failure, brittle failure, cycle fatigue, thermal stratification and aging. The critical cracks were calculated conservatively in consideration of the bounding material properties and load conditions in the load limit and fracture mechanics analyses. The licensee's LBB methods are consistent with the guidance in SRP Section 3.6.3 and the potential for fatigue crack growth in the subject piping is insignificant and does not affect the crack stability and the validity of the LBB analysis. In addition, the NRC staff finds that the licensee's analysis has demonstrated that the subject piping has an extremely low probability of rupture.

In addition, based on its review of the information provided in the LAR, as supplemented, and the NRC staff's previous review and approval of PCS leakage detection system as a part of PNP License Amendment No. 162, the staff finds that the PCS leakage detection system conforms to the guidelines provided by RG 1.45, Revision 1, and is therefore, acceptable.

Pursuant to 10 CFR Part 50, Appendix A, GDC 4, the NRC staff concludes that the licensee is permitted to exclude consideration of the dynamic effects associated with the postulated rupture of the subject piping from current licensing basis at PNP.

On July 24, 2025, the NRC staff issued its approval of an exemption request (ML25163A182) to allow for a one-time rescission of the PNP docketed 10 CFR 50.82(a)(1) certifications to remove the restriction that prohibits operation of the PNP reactor and emplacement and retention of fuel into the PNP reactor vessel. The exemption will take effect on August 25, 2025, Holtec's planned date to transition to the power operations licensing basis. As such, this license amendment is effective upon the licensee's submittal of a request to rescind the 10 CFR 50.82(a)(1) certifications.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations on, the Michigan State official was notified of the proposed issuance of the amendment on July 29, 2025. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

In accordance with 10 CFR 51.30, 10 CFR 51.31, and 10 CFR 51.32, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment, as discussed in the NRC staff's environmental assessment and finding of no significant impact, issued on May 30, 2025 (90 FR 23071).

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

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Date of Issuance: August 26, 2025

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT NO. 280 RE: LEAK-BEFORE-BREAK METHODOLOGY FOR PRIMARY COOLANT SYSTEM HOT AND COLD LEG PIPING (EPID L-2025-LLA-0027) DATED AUGUST 26, 2025

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