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

October 30, 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. 281 REGARDING A CHANGE TO TECHNICAL SPECIFICATIONS TO ALLOW REPAIRING STEAM GENERATOR TUBES BY SLEEVING (EPID L-2025-LLA-0036)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant. The amendment consists of changes to the license in response to your application dated February 11, 2025, as supplemented by letters dated May 29, 2025, July 30, 2025, August 29, 2025, September 23, 2025, and October 7, 2025, respectively.

The amendment revises the technical specifications to allow the use of Framatome Alloy 690 sleeves to repair defective steam generator tubes as an alternative to removing the tubes from service by plugging.

The NRC has determined that the related safety evaluation contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, requests for withholding." The proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a non-proprietary publicly available version of the safety evaluation, which is provided as Enclosure 2. The proprietary version of the safety evaluation is provided as Enclosure 3.

Enclosure 3 to this letter contains proprietary information. When separated from Enclosure 3, this document is DECONTROLLED.

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. 281 to DPR-20
2. Safety Evaluation (Non-Proprietary)
3. Safety Evaluation (Proprietary)

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. 281
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, dated February 11, 2025, as supplemented by letters dated May 29, 2025, July 30, 2025, August 29, 2025, September 23, 2025, and October 7, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment, and paragraphs 2.C.(2), 2.K, 2.L, and 2.M of Renewed Facility Operating License No. DPR-20 are hereby amended to read as follows:

   (2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Palisades Energy shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   K. Palisades Energy will inspect 100 percent of the steam generator in-service sleeve/tube assemblies by the end of cycle 29 refueling outage (1R29).

   L. Palisades Energy will perform an inspection of a minimum of 50 percent of all SG in-service tube/sleeve assemblies by the end of each refueling outage after 1R29. In the initial 50 percent sample, if a flaw is identified in the four expansion areas at either end of the sleeve of an in-service tube/sleeve assembly (A600 parent tube or A690 sleeve) or in the A690 sleeve between the expansion areas at the ends of the sleeve, a scope expansion will be applied to the remaining 50 percent of the in-service tube/sleeve assemblies. Identifying a flaw in the non-pressure boundary portion of the A600 parent tubing (behind the center portion of the sleeve) will not require a scope expansion. The Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines (ID 3002020909) will be used to select the minimum expansion scope.

   M. This license is effective as of the date of issuance and shall expire at midnight March 24, 2031.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-20

Date of Issuance: October 30, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 281

PALISADES NUCLEAR PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Renewed Facility Operating License No. DPR-20

Replace the following pages of the Renewed Facility Operating License No. DPR-20 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating areas of change.

REMOVE	INSERT
3	3
8	8
-----------	9

Technical Specifications

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

REMOVE	INSERT
3.4.1-2	3.4.1-2
3.4.17-1	3.4.17-1
3.4.17-2	3.4.17-2
5.0-10	5.0-10
5.0-11	5.0-11
5.0-12	5.0-12
5.0-13	5.0-13
5.0-14	5.0-14
5.0-15	5.0-15
5.0-25	5.0-25
5.0-26	5.0-26

(2) Palisades Energy, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) Palisades Energy, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Palisades Energy is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Palisades Energy shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Fire Protection</u>

Palisades Energy shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment requests dated December 12, 2012, November 1, 2017, November 1, 2018, and March 8, 2019, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014, August 14, 2014, November 4, 2014, December 18, 2014, January 24, 2018, and May 28, 2019, as approved in the safety evaluations dated February 27, 2015, February 27, 2018, and August 20, 2019. Except where NRC approval for changes or

H. The Updated Safety Analysis Report supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the Updated Safety Analysis Report required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, Palisades Energy may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Palisades Energy evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

I. The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Palisades Energy shall complete these activities no later than March 24, 2011, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

J. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal scheduled, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

K. Palisades Energy will inspect 100 percent of the steam generator in-service sleeve/tube assemblies by the end of cycle 29 refueling outage (1R29).

L. Palisades Energy will perform an inspection of a minimum of 50 percent of all SG in-service tube/sleeve assemblies by the end of each refueling outage after 1R29. In the initial 50 percent sample, if a flaw is identified in the four expansion areas at either end of the sleeve of an in-service tube/sleeve assembly (A600 parent tube or A690 sleeve) or in the A690 sleeve between the expansion areas at the ends of the sleeve, a scope expansion will be applied to the remaining 50 percent of the in-service tube/sleeve assemblies. Identifying a flaw in the non-pressure boundary portion of the A600 parent tubing (behind the center portion of the sleeve) will not require a scope expansion. The Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines (ID 3002020909) will be used to select the minimum expansion scope.

M. This license is effective as of the date of issuance and shall expire at midnight March 24, 2031.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A –Technical Specifications

2. Appendix B – Environmental Protection Plan

Date of Issuance: January 17, 2007

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify PCS cold leg temperature within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	------------------------------NOTE---------------------------- Not required to be performed until 31 EFPD after THERMAL POWER is ≥ 90% RTP. --- Verify PCS total flow rate within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program <u>AND</u> After each plugging or repairing (or both) of the number of steam generator tubes which results in the same primary flow reduction as plugging 10 or more steam generator tubes

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging or repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging or repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging or repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.7 (Deleted)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5.8 Steam Generator (SG) Program (Continued)

b. Performance criteria for SG tube integrity. (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS Operational LEAKAGE."

c. Provisions for SG tube plugging or tube repair criteria. Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired. Sleeved tubes found by inservice inspection to contain a flaw in a sleeve or a flaw in the sleeve to tube joint shall be plugged. The following SG alternate repair criteria may be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service induced flaws between the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, and 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, shall be plugged. Flaws located below 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, may remain in service.

2. Tubes found by inservice inspection to contain service induced flaws between the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, may remain in service.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 12.5 inches below the

5.5.8 Steam Generator (SG) Program (Continued)

d. Provisions for SG tube inspections. (continued)

bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and that may satisfy the applicable tube plugging or tube repair criteria. The tube to tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

4. When the SG alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the SG alternate repair criteria of TS 5.5.8c.1 every 24 effective full power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.8 Steam Generator (SG) Program (Continued)

f. Provisions for SG tube repair methods. SG tube repair methods shall provide the means to reestablish the PCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a tube repair. All acceptable tube repair methods are listed below.

1. Framatome Document Number 51-9385467-002, "Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant." The sleeve shall remain in service for no more than ten years of operation starting from the outage when the sleeve was installed.

5.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation (FHAV) systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (Continued)

a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
FHAV (single fan operation)	7300 ± 20%
FHAV (dual fan operation)	10,000 ± 20%
CRV	3,200 +10% -5%

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV system and < 1.00% for the FHAV system when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
FHAV (dual fan operation)	10,000 ± 20%
CRV	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
FHAV	6.00%	95%
CRV	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
FHAV (dual fan operation)	6.0	10,000 ± 20%
CRV	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
CRV	15 kW

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (Continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

5.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

    1. API gravity or an absolute specific gravity,

    2. Kinematic viscosity, and

    3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

    1. A change in the TS incorporated in the license; or

    2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.6 Reporting Requirements

5.6.5 COLR (Continued)

21. BAW-10240(P)-A, "Incorporation of M5 Properties in Framatome ANP Approved Methods." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism, 5.6 Reporting Requirements

5.6.8 Steam Generator Tube Inspection Report (continued)

f. Total number and percentage of tubes plugged or repaired to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The effective plugging percentage for all tubes plugged or repaired in each SG,

i The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided,

j. The tube repair methods utilized and the number of tubes repaired by each repair method.

ENCLOSURE 2

NON-PROPRIETARY SAFETY EVALUATION

RELATED TO AMENDMENT NO. 281 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-20

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.
Redacted information is identified by blank space enclosed within [[double brackets]].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 281 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 11, 2025 (Reference 1), as supplemented by letters dated May 29, 2025 (Reference 2), July 30, 2025 (Reference 3), August 29, 2025 (Reference 4), September 23, 2025 (Reference 5), and October 7, 2025 (Reference 6), Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Palisades, LLC,[2] (collectively, Holtec) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC or Commission) 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 Technical Specifications (TS) to allow the use of Framatome Alloy 690 sleeves to repair defective steam generator (SG) tubes as an alternative to removing the tubes from service by plugging. The approval of this license amendment request (LAR) was contingent upon the approval of the LAR dated December 14, 2023 (Reference 7), to reflect the resumption of power operations at PNP. The NRC approved that amendment on July 24, 2025 (Reference 8).

The supplements dated May 29, 2025, July 30, 2025, August 29, 2025, September 23, 2025, and October 7, 2025, provided additional information that clarified the application, did not expand the scope of the application as originally noticFed, 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 15722). Please note that the staff's safety evaluation (SE) contains proprietary information and is thus marked accordingly with double square brackets (**[[ ]]**).

2 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.

1.1 Background

By letter dated January 4, 2017 (Reference 9), pursuant to Paragraph (a)(1)(i) of Section 50.82, "Termination of license," of Title 10 of the Code of *Federal Regulations* (10 CFR), Entergy Nuclear Operations, Inc. (Entergy), the previous licensee for PNP, certified to the NRC that it decided to permanently cease power operations at PNP by October 1, 2018. By letters dated September 28, 2017 (Reference 10), and October 19, 2017 (Reference 11), Entergy updated its timeline and certified to the NRC that it planned to permanently cease power operations at PNP no later than May 31, 2022. By application dated December 23, 2020 (Reference 12), as supplemented, Entergy on behalf of itself, Entergy Nuclear Palisades, LLC, Holtec International, and HDI submitted a license transfer application to transfer the PNP license from Entergy to Holtec. By letter dated December 13, 2021 (Reference 13), the NRC issued an order consenting to the license transfer.

On May 20, 2022, PNP permanently ceased power operations. Pursuant to 10 CFR 50.82(a)(1)(ii), by letter dated June 13, 2022 (Reference 14), Entergy certified to the NRC that all fuel had been permanently removed from the PNP reactor vessel and placed in the spent fuel pool on June 10, 2022. These certifications were docketed by the NRC. Upon docketing the 10 CFR 50.82(a)(1) certifications, 10 CFR 50.82(a)(2) no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel. Shortly after PNP transitioned to a permanently shut down and defueled facility in accordance with 10 CFR 50.82(a)(2), Holtec Palisades, LLC assumed ownership of PNP, and HDI became the licensed operator for decommissioning PNP (Reference 15) and began the decommissioning process.

In early 2023, HDI engaged with the NRC staff regarding the potential restart of reactor operation at PNP. By letter dated March 13, 2023 (Reference 16), HDI submitted its proposed regulatory path to reinstate the power operations licensing basis to resume power operations at PNP through a series of licensing submittals referred to as a "regulatory framework." Holtec's proposed regulatory framework consisted of an exemption from 10 CFR 50.82(a)(2) and a series of licensing actions to restore the plant's licensing basis to the one in effect just prior to permanent shut down.

Specifically, one of the proposed licensing actions, which was submitted on December 14, 2023, was a LAR in support of resuming power operations that largely seeks to undo the changes made by the previously issued Permanently Defueled TS Amendment. Consistent with supporting the resumption of power operations, the LAR dated December 14, 2023, proposed to reinstate TSs associated with the PNP SGs that were in effect just prior to shutting down the PNP reactor. The PNP TSs, as provided in the LAR dated December 14, 2023, allowed defective SG tubes to be removed from service by installing plugs at both ends of a tube, which removes the heat transfer surface of the plugged tube from service. The NRC staff approved that LAR on July 24, 2025.

Because PNP was permanently shutting down, the SG tube inspections normally conducted during refueling outages were not conducted after operating cycle 28, in May of 2022. When SG tube inspections were previously completed in the fall of 2020, during refueling outage 27 (RFO 27), 63 tubes were plugged (36 in SG A and 27 in SG B), due to various degradation mechanisms such as stress corrosion cracking (SCC) and tube wear. Details are contained in the SG tube inspection report dated March 25, 2021 (Reference 17). After RFO 27, the total number of plugged tubes was 1,159 (666 in SG A and 493 in SG B). Prior to the installation of the SGs, Combustion Engineering had advised Consumers Energy (the then-current licensee) that the area around the center stay cylinder region was potentially susceptible to fretting wear at the diagonal straps (also called “bat wings”). As a result, 308 tubes in SG A and 309 tubes in SG B were preventively plugged before the PNP SGs were placed into service (Reference 18). Therefore, at the time of shutdown in May 2022, a total of 542 tubes had been plugged during PNP operation (358 in SG A and 184 in SG B).

While PNP was in a decommissioned status, the licensee performed SG tube inspections as part of the effort to restart power operations. On September 3, 2024, the NRC staff and representatives from Holtec participated in a conference call to discuss the ongoing SG tube inspection activities at PNP (Reference 19). During the call, Holtec presented preliminary information that indicated an additional 701 tubes in the two SGs were candidates for plugging. However, plugging these 701 tubes would exceed the design plugging limit for SG A. Consequently, this LAR was submitted to allow the installation of sleeves in the tubes of both SGs that would maintain the total number of tubes plugged in SG A within the plugging limit and keep additional tubes in service in both SGs.

The changes proposed in the LAR dated February 11, 2025, which is the subject of this SE, revised the affected TS and TS Bases pages included in the LAR dated December 14, 2023, to permit defective SG tubes to be repaired using Alloy 690 sleeves. The LAR dated February 11, 2025, would permit plugging or repair of SG tubes, as appropriate, prior to the restart of the unit and during subsequent outages.

Note, 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 the LAR, RCS is used interchangeably with PCS and RCPB is used interchangeably with PCPB, depending on the document being referenced.

1.2 Proposed Changes to PNP Technical Specifications

The following is a description of the PNP TS changes proposed by this LAR, as supplemented, to include the option to repair SG tubes:

- TS 3.4.1, “PCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits,” Surveillance Requirement (SR) 3.4.1.3 is revised to require the PCS total flow rate to be verified within limits whenever plugging or repairing (or both) of the SG tubes results in the same primary flow reduction as plugging 10 or more SG tubes.

- TS 3.4.17, “Steam Generator (SG) Tube Integrity,” is revised by adding the option to repair SG tubes to limiting condition for operation (LCO) 3.4.17, associated remedial action (Condition A and Required Action A.2), and SR 3.4.17.2. The current TS 3.4.17 only allows tube plugging.

- TS 5.5.8, “Steam Generator (SG) Program,” is revised to allow the option to repair SG tubes, update the provisions for SG tube repair criteria, and to specify that the allowable SG tube repair method is a sleeve. The sleeve shall remain in service for no more than ten years of operation.

- TS 5.6.8, Steam Generator Tube Inspection Report, is revised to add reporting requirements for repaired tubes and to correct editorial errors.

2.0 REGULATORY EVALUATION

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

2.1 System Description

In its LAR, dated February 11, 2025, Section 3.2, "System Description," Holtec described the SGs installed at PNP as follows:

> The PNP PCS is comprised of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one SG and two circulating pumps. The SG operates with the primary coolant in the tube side and the secondary fluid in the shell side.
>
> The PNP SGs are vertical shell and "U" tube units. The primary coolant pressure boundary (tube side) of the SGs is designed to ASME [American Society of Mechanical Engineers] Section III Class 1 rules. Critical components of the Class 2-design secondary side (e.g., the feedwater nozzles) were also analyzed using Class 1 methods.
>
> The design parameters for the SGs are provided in Updated Final Safety Analysis Report (UFSAR) Revision 35, Table 4-4, Steam Generator Parameters (ML21125A3008). The SGs are designed to withstand the pressure differential of PCS operating pressure and atmospheric pressure. Note that all PNP UFSAR references refer to Revision 35 as it describes the PNP power operations licensing basis (POLB), which will be restored to support the resumption of power operations. Additional details of the PNP SGs are provided in Section 3.4.2 of this [LAR] Enclosure and UFSAR Section 4.3.4, *Steam Generator*.

The PNP has two Combustion Engineering Model 2530 replacement SGs. Each SG has 8,219 mill-annealed Alloy 600 tubes. The tubes have a nominal outside diameter of 0.75 inches, and a nominal wall thickness of 0.042 inches. Stainless steel lattice-type tube supports, diagonal straps, and vertical straps support the tubes at various locations.

2.2 Applicable Regulatory Requirements and Guidance

Regulations in 10 CFR Section 50.90, "Application for amendment of license, construction permit, or early site permit," requires in part, that whenever a licensee desires to amend the license, an application for an amendment must be filed with the Commission fully describing the changes desired and following, as far as applicable, the form prescribed for original applications.

Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for licenses and construction permits in 10 CFR 50.40(a) (regarding, among other things, consideration of the operating procedures, the facility and equipment, the use of the facility, and other TSs, or the proposals), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be “reasonable assurance” that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission's regulations.

Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 requires, in part, a quality assurance program for the design, fabrication, construction, and operation of structures, systems, and components in nuclear plants. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

The SG tubes are part of the RCPB and isolate fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this SE, SG tube integrity means that the tubes can perform this safety function in accordance with the plant design and licensing basis. The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements that state that the RCPB shall have “an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture” (GDC 14), shall be designed with sufficient margin” (GDCs 15 and 31), shall be of “the highest quality standards practical” (GDC 30), and shall be designed to permit “periodic inspection and testing...to assess...structural and leaktight integrity” (GDC 32). Palisades received a construction permit prior to July 7,1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. The licensee states in the LAR that it complies with the intent of the current GDC and meets the principal design criteria that were in effect when PNP was licensed. Section 4.5 of the LAR discusses how the licensee meets the principal design criteria for PNP in UFSAR Sections: 5.1.3.5 (GDC 14), 5.1.3.6 and 5.2 (GDC 15), 5.1.5.1 (GDC 30), 5.1.5.2 (GDC 31), and 5.1.5.3 (GDC 32).

As part of the plant’s licensing bases, applicants for pressurized water reactor (PWR) licenses are required to analyze the consequences of postulated design-basis accidents, such as a SG tube rupture and a main steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100 accident source term, GDC 19 for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC approved licensing basis (e.g., a small fraction of these limits). The LAR also states how the licensee meets the control room dose limit requirements of GDC 19. No accident analyses for PNP are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The use of the proposed sleeving repair reinforces the integrity of the underlying SG tubes, and the SG tubes would meet the intent of the requirements of the GDC in Appendix A to 10 CFR Part 50, as discussed below. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

Section 50.55a(c)(1) of 10 CFR requires, in part, that reactor coolant pressure boundary components meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Code. Because PNP was issued a construction permit prior to 1971, Section 50.55a(g)(1) further requires, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, ASME Code Class 1 components must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” of the ASME Code, to the extent practical. Therefore, the SG sleeve repair method must be qualified in accordance with Section XI of the ASME Code for inspection, and must meet ASME Code, Section III limits for design, operating conditions, and accident loading conditions.

Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes” (Reference 20), provides guidance for determining the minimum SG tube wall thickness and for determining the repair criteria for SG tubes with sleeves. In accordance with RG 1.121, the margin of safety against tube rupture under normal operating conditions should not be less than three at any tube location where flaws have been detected. The margin of safety against tube failure under postulated accidents, such as a loss-of-coolant accident, main steam line break, or feed water line break concurrent with the safe shutdown earthquake, should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Code.

The NRC staff’s guidance for the review of operational TS is contained in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Chapter 16.0, “Technical Specifications,” Revision 3, dated March 2010 (Reference 21). As described therein, as part of a regulatory standardization effort, the NRC staff has prepared standardized technical specifications (STS) for each of the LWR nuclear designs. The NRC staff approved PNP’s TS conversion to the STS in November 1999 (Reference 22). The latest STS revision applicable to PNP is NUREG-1432, Revision 5.0, “Standard Technical Specifications, Combustion Engineering Plants,” Volume 1, “Specifications,” and Volume 2, “Bases,” dated September 2021 (Reference 23).

The regulation in 10 CFR 50.36, “Technical specifications,” establishes the requirements related to the content of TS. Pursuant to 10 CFR 50.36(c)(1)–(5), TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant’s TSs.

The regulation in 10 CFR 50.36(c)(2)(i) requires TS to include items in the “Limiting conditions for operation” category stating in part:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulation in 10 CFR 50.36(c)(3) requires TS to include items in the “Surveillance requirements” category stating:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulation in 10 CFR 50.36(c)(5) requires TS to include items in the "Administrative Controls" category, which are "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." Programs established by the licensee, including the SG program, are listed in the administrative controls section of the TSs to assure operation of the facility in a safe manner. For PNP*,* the requirements for performing SG tube inspections and repair are in TS 5.5.8, "Steam Generator (SG) Program," while the reporting requirements for SG tube inspections and repair are in TS 5.6.8, "Steam Generator Tube Inspection Report."

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine whether the proposed changes are consistent with the regulatory requirements, guidance, and licensing and design-basis information discussed in Section 2.2 of this SE.

3.1 TS Requirement for SG Tube Integrity

The TSs for all PWR plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For PNP, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.8.b. for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.8.a. requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. Technical Specification 5.5.8.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube repair criteria. The applicable SG tube repair criteria, specified in TS 5.5.8.c are that tubes found during in-service inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of alternate repair criteria provided in TS 5.5.8.c.1 or 5.5.8.c.2. The provisions for the sleeving repair method are contained in proposed TS 5.5.8.f.1.

3.2 Sleeve Background and Operational Experience

In support of this LAR, Holtec submitted Framatome Document Number 51-9388710-001, *"Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant"* (Reference 24). Reference 24 states that Framatome U.S. has installed 23,933 sleeves in SGs and other heat exchangers. Between 1978 and 1997, Framatome U.S. installed 10,682 sleeves in both recirculating and once-through SG designs in the United States and Europe; however, none of the 10,682 SG sleeves are currently in service because those SGs were either replaced, or the plant was retired. The sleeve proposed for installation in the Palisades SGs was designed, developed, and qualified in the 1994 timeframe, just as the market for installing SG sleeving was shrinking due to SG replacements. Framatome states that it has installed an additional 13,251 sleeves to date in other heat exchangers such as feedwater heaters, component cooling water (CCW) heat exchangers, containment spray heat exchangers, and reactor building spray heat exchangers. The sleeve design being proposed for Palisades is referred to as the H8 sleeve, since it contains eight hydraulically expanded regions. The H8 sleeve has been installed 6,007 times in non-SG heat exchangers. Of these 6,007 installations, the H8 design has been installed 964 times in ASME Code Section III heat exchangers, including CCW and containment spray heat exchangers. The sleeves installed in the ASME Code Section III heat exchangers fall under 10 CFR Part 21 Reporting of Defects and Noncompliance and Framatome reported that it has received no 10 CFR Part 21 notifications pertaining to these sleeves. Framatome also stated that it is not aware of any issues with the H8 sleeves installed in the non-SG applications.

Leak-limiting sleeves made from Alloy 800 and designed by Westinghouse, which have some similar features to the Framatome H8 sleeves, were approved for use and installed in the original SGs at Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Watts Bar Nuclear Plant, Unit 1, St. Lucie Plant, Unit No. 2, Comanche Peak Nuclear Power Plant, Unit No. 1 (Reference 25). The leak-limiting Alloy 800 sleeve was approved for use at Watts Bar, Unit 2, but no sleeves were installed at Watts Bar, Unit 2, prior to SG replacement.

In addition, Westinghouse leak-limiting Alloy 800 sleeves have been approved for both support plate and tubesheet installations at Beaver Valley, Unit 2, but only tubesheet sleeves have been installed. The Westinghouse tubesheet sleeves were first installed in the Beaver Valley, Unit 2, SGs in 2012, and as of 2025, there are 1,012 tubesheet sleeves in service (Reference 26). Except for the SGs at Beaver Valley, Unit No. 2, all other units have replaced their SGs. Additionally, thousands of Alloy 800 leak-limiting sleeves have been installed in nuclear plants outside the U.S (Reference 25).

3.3 Evaluation of the Proposed Use of Leak-Limiting Sleeves

The licensee has proposed a tube repair method that uses a sleeve to repair SG tubes with flaws, as an alternative to removing the tubes from service by plugging. The design, installation, and qualification tests of the sleeve are documented in Reference 24. On May 20 and 21, 2025, the NRC staff performed an onsite audit of Framatome's H8 sleeve testing and qualification program (Reference 27). In response to a request for additional information (RAI) from the NRC staff, the licensee submitted References 28-34. The NRC staff's review of the sleeve design, installation, and qualification is documented below.

3.3.1 Sleeve Design

A sleeve is a tube segment with a slightly smaller diameter than a SG tube, which is inserted into a SG tube and hydraulically expanded at both ends to form a structural joint via an interference fit. The sleeve design being proposed for use at Palisades is a leak-limiting sleeve design. Each H8 sleeve contains eight hydraulic expansion areas, four near each end of the sleeve, that create an interference fit of a sleeve within a SG tube. The sleeve is fabricated with multiple sealing ribs on the outer diameter of all eight hydraulic expansion areas. The profile of these ribs, when combined with the sleeve expansions, improves the sleeve's leak-limiting characteristics. The sleeves are designed for use at locations in SG tubes where the tubes pass through the lattice-type tube supports.

Defects in the parent tube at the lattice support location are repaired by sleeve installation since the new primary-to-secondary pressure boundary at the support location is the Alloy 690 sleeve, not the parent tube. The licensee stated that the sleeve design calculations use allowable stresses based on the strength properties listed in ASME Boiler and Pressure Vessel Code (ASME Code) Section II, for Alloy 600 (for the tubes) and Alloy 690 (for the sleeves). The sleeves are constructed in accordance with ASME Code Section III, Subsection NB and that the design and installation procedure was qualified by testing for leakage in accordance with ASME Code Section XI, IWA-4721 and IWA-4725. Additional discussion about ASME Code qualification testing is provided in Sections 3.3.4 and 3.3.5 of this SE.

3.3.2 Sleeve Materials Selection

The sleeve material is Alloy 690, which is a nickel-chromium-iron alloy that is used extensively in the nuclear industry for its favorable material properties, including corrosion resistance in both the primary and secondary side water chemistries. As of September 2025, 46 U.S. PWRs use Alloy 690 SG tubing, with the oldest unit in operation since 1989, and SCC has never been detected in the Alloy 690 SG tubing fleet. The Alloy 690 material was procured in accordance with the requirements of the ASME Code, Section II, which is acceptable to the NRC staff.

3.3.3 Tube Inspections and Sleeve Installation

The licensee stated in the LAR that installation of the H8 sleeve is accomplished remotely by tooling attachments mounted on a manipulator. The NRC staff observed the installation of an H8 sleeve in a tube during an NRC audit at Framatome in Lynchburg, VA in May 2025. The sleeve installation and inspection methods used minimize the personnel radiation exposures in accordance with ALARA (as low as reasonably achievable) principles. The pre-installation inspections include eddy current (EC) examination of the tubes to be sleeved, to verify there is no tube degradation in the regions where the upper and lower hydraulic expansions will be created. The distances from the lattice-grid supports to the diagonal supports are confirmed by EC examination (if not already known) to ensure sleeves are not installed adjacent to diagonal supports or other external structural obstructions. In addition, any dents or dings observed during EC examination in a target tube are noted for further testing, which will determine if the dent or ding could inhibit sleeve installation. The sleeves are installed in accordance with the sleeving procedure specification that was developed during the sleeve qualification process (Reference 35). A total of eight joints are made simultaneously, four above and four below the tube support location. After the eight hydraulic expansions are made, a partial expansion is made in the center of the sleeve with a different expansion tool. The center expansion is used to preload the upper and lower expansion joints. After sleeve installation, an EC examination verifies the location of the sleeve relative to the tube degradation and establishes a baseline inspection of the sleeve and tube that allows for future comparisons of the sleeve and tube conditions. Sleeves that are found to be installed offset from the intended location undergo engineering disposition to determine whether they can remain in service.

3.3.4 Sleeve Qualification Testing

Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,* to 10 CFR Part 50 requires, in part, a quality assurance program for the design, fabrication, construction, and operation of structures, systems, and components in nuclear plants. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. The PNP operational and support activities quality assurance program is discussed in Section 15.1 of the UFSAR, Revision 35. The NRC staff reviewed the quality assurance program outlined in Section 15.1 of the UFSAR, Revision 35, and confirmed that it addresses corrective actions, a confirmation process, and administrative controls, which meets the requirements for a pre-GDC plant such as Palisades.

The sleeve qualification considered temperatures, pressures, and cyclic testing that bound plants having steam generators with ¾-inch tubing. The Palisades sleeve design specified loading, geometry, and materials specific to the Palisades SGs. The materials, design, fabrication, and examination of the Palisades SG tubes meet all the requirements of ASME Code, Section III, 1977 Edition. The licensee performed a reconciliation to demonstrate that the qualification testing and analysis for the sleeves used the same or bounding loading, geometry and materials specified in the design specification. The sleeve repair activities are completed in accordance with the NRC approved version of the ASME Code, Section XI, 2007 Edition with Addenda through 2008, which is the Palisades approved code of record. The sleeves supplied by Framatome were originally qualified to the NRC approved version of ASME Section XI, 1989 Edition with no Addenda and supplemented with ASME Section XI, 2007 Edition with Addenda through 2008. The relevant paragraph in Section XI, 1989 Edition with no Addenda used to qualify the sleeve was IWB-4300, *Heat Exchanger Tube Sleeving.* The relevant sections in the 2007 Edition with 2008 Addenda of the Code are IWA-4720, *Sleeving*, IWA-4721, *General Requirements*, and IWA-4725, *Expansion*. The NRC staff reviewed the comparison between the ASME Code editions and found only administrative changes. There were no changes to technical requirements between the ASME Code editions. Therefore, the technical requirements of the ASME Code, Section XI, 2007 Edition with Addenda through 2008 are met and are acceptable to the NRC staff. (Note: Use of a later code is permitted by ASME Code, Section XI, IWA-4221, provided reconciliation is performed in accordance with IWA-4222 through IWA-4226, as applicable. Those paragraphs require the sleeves to be evaluated for changes to the weight, configuration, and pressure-temperature rating of the SGs.)

3.3.5 ASME Section XI Mechanical Testing of the Sleeves

ASME Code, Section XI, IWA-4725.2.1(d) specifies the qualification test requirements for hydraulically expanded sleeves. Test load ranges and required cycles are obtained per ASME Code Section III, Appendix II, and are based on the number of test assemblies and various factors relating the test conditions to the actual plant operating conditions. The test loads were originally calculated for the design conditions and transients of certain Westinghouse model SGs, which bound, and are therefore conservative for, the fatigue loadings applicable to the H8 sleeves being installed at PNP.

Per IWA-4725.2.1(d), fatigue testing is required to demonstrate that the sleeve attachment can withstand the specified design loadings per ASME Code Section III, Appendix II (Experimental Stress Analysis) without exceeding the specified design leakage limit. In addition to the fatigue cycling, thermal and pressure cycling was performed, as well as room temperature and operating temperature leak tests. Details are provided below.

3.3.5.1 Specimens

The mechanical test specimens are described in Section 8.1 of Enclosures 5 and 5a to Reference 24. The sleeves were installed using the Framatome hydraulic expansion system. A range of tube outside diameter (OD) expansion sizes were produced during testing to simulate possible SG conditions. After the eight expansions were made, a partial expansion was placed in the center of the sleeve to preload the upper and lower set of hydraulic joints. This preload allows the hydraulic expansion joints between the sleeve and tube to more effectively reduce leakage. Some of the specimens were fabricated with intentional defects and tested to demonstrate that a single joint could carry the operating and accident loads. In addition, the tubing qualification tests used a range of yield strengths that bound the range of tubing yield strength in the PNP SGs.

3.3.5.2 Leak Testing

The specimens received either an initial room temperature leak test or a no-load room temperature leak test. Typically, the leak rates for these specimens remained relatively unchanged as testing progressed. Operating temperature leak testing was also performed before and after cycling on various specimens during the qualification process.

3.3.5.3 Cyclic Testing

The adequacy of the sleeve and its joints to withstand operational loading was demonstrated by subjecting multiple specimens to axial, thermal, and pressure cycling. These tests were performed to expose the joints to the conditions that the sleeves would be subjected to in the SG, to determine if the plant operating conditions would degrade the leakage performance of the sleeve joint. The multiple specimens were axially cycled to represent 10 years of structural design life, and none of the specimens recorded sleeve joint motion. The multiple specimens were also thermally cycled to determine if differences in the thermal expansion coefficients for the sleeve and tube would affect the hydraulic expansion joints. None of the samples recorded sleeve motion during these tests. Pressure cycling was also performed on multiple specimens. This test represented the startup and shutdown pressure cycling to which a sleeve will be exposed during 10 years of design life, and no sleeve-to-tube movement was noted during testing.

3.3.5.4 Main Steam Line Break Testing

After the leak tests and cyclic tests were complete, the sample pressures were increased to a minimum of 2780 pounds per square inch gauge, with minimum hold times to represent the maximum pressure expected from a main steam line break (MSLB) accident. The leak rates measured during this MSLB testing, performed at room temperature, were higher than those previously recorded during room temperature leak testing at lower pressure. However, no joint failure was noted during testing.

3.3.5.5 Sleeve Leakage Integrity

Results from laboratory testing show that potential leakage past the hydraulic expansions is very low and well below the maximum allowable limits in the plant's TS. The operational leakage integrity of degraded tubes is assessed using the criteria outlined/defined in the performance criteria contained in Section 2.3 of the Steam Generator Program Guidelines, Nuclear Energy

Institute (NEI) 97-06 (Reference 28). The PNP TS 3.4.13 includes a limit of 150 gallons per day (gpd) on primary-to-secondary operational leakage through any one SG, beyond which the plant must be promptly shutdown; however, PNP has adopted an administrative limit of 72 gpd on primary-to-secondary operational leakage. Should a flaw exceeding the tube repair limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural, and accident induced leakage integrity are impaired. The operational leakage limit is monitored through plant equipment.

For MSLB leakage calculations, each installed sleeve is assumed to be leaking at the MSLB leakage rate. Therefore, the MSLB leakage value will limit the number of sleeves in each SG that can be installed at PNP since the accident induced leakage limits assure that any postulated leakage in the limiting SG will be less than the bounding condition necessary to ensure that offsite doses during an accident remain a small fraction of the 10 CFR Part 100 site criteria and control room dose is less than the 10 CFR 50 Appendix A, GDC 19 limits.

3.3.5.6 Mechanical Testing Summary

Mechanical tests were performed on sample sleeved tubes to determine leak rate properties. These tests included initial leak rates, axial, thermal and pressure cycling, and final leak rates at normal operating and MSLB conditions. The results demonstrated the axial load carrying capability of the sleeves is sufficient (i.e., provide an adequate safety factor for normal and postulated accident conditions). Mechanical testing also determined that the installed sleeve will withstand the cyclic loading resulting from power changes and other transients in the plant. The cyclic testing duration limits the amount of time that the sleeves will be in service to 10 years. The NRC staff finds the mechanical test results acceptable.

3.3.6 Corrosion Evaluation

Since Alloy 690 has demonstrated superior resistance to SCC in both laboratory testing and in SG operating experience relative to Alloy 600, the Framatome investigation of the effect of residual stress from sleeve installation focused on the Alloy 600 parent tubing. Section 6 in the LAR discusses various analyses and corrosion tests. X-ray diffraction and finite element analysis (FEA) were performed following hydraulic expansion, to identify the location of the peak residual stresses. These results from sleeve expansion were found to agree well with industry evaluation of hydraulic and explosive tube expansions into the tubesheet. A stress indexing test was performed to identify locations that could be susceptible to SCC. In response to an NRC staff question, the licensee identified and corrected a typographical error in the stress indexing crack location. The crack locations in the stress indexing tests were consistent with the predicted peak tensile stress location from X-ray diffraction and FEA results. Various caustic corrosion tests were performed on samples at elevated temperatures to assess the relative susceptibility to SCC. The residual stress and corrosion test results were then used to project when primary water stress corrosion cracking would be expected in the parent tubing at the hydraulic expansions formed during sleeve installation.

3.3.7 Sleeve and Tube Eddy Current Inspection Post Installation

Since the Alloy 690 sleeves are being installed into mill-annealed Alloy 600 tubing at PNP, an important part of the NRC staff review was focused on the ability to detect flaws in the Alloy 600 parent tubing, should SCC occur after the sleeves are in service.

After sleeve installation, a bobbin coil EC inspection verifies the location of the sleeve relative to the tube degradation at the lattice tube support. After the bobbin coil inspection, a “sleeve” probe inspection establishes a baseline inspection of the sleeve and tube that allows for future comparisons of the sleeve and tube conditions. If service-related degradation is detected in any portion of the sleeve or sleeve-to-tube joint during future inspections, the SG tube will be taken out of service by installing site-approved, mechanical plugs at both ends of the tube. Flaws in the parent tube at the lattice tube supports located adjacent to the center portion of the sleeve do not result in tube plugging, since the parent tube in this location is no longer a pressure boundary. Unsleeved portions of a sleeved tube will continue to use the PNP standard 40 percent through-wall (TW) TS plugging limit.

To ensure that effective inspections of the sleeve/tube assembly could be performed, the capability to inspect these regions was assessed with a specialized EC +Point™ sleeve probe. This probe was specifically designed for traversing the standard diameter SG tubing, the narrower diameter sleeving, and the transition from the sleeve to the parent tubing with a frequency optimized for inspection of hydraulically expanded sleeves. The qualification of the sleeve probe was performed in accordance with the requirements contained in Appendix H of the Electric Power Research Institute (EPRI) Technical Report No. 3002007572, *Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 8; Final Report June 2016 with Interim Guidance Incorporated, April 2019 (SGMP-19-01) and September 2021 (SGMP-21-02)* (Reference 36).

The initial qualification program included fabricating samples of sleeve/tube assemblies with axially and circumferentially oriented flaws on the inside diameter (ID) and OD of the parent tube and of the sleeve. These flaws were notches of various lengths and depths, created by electro discharge machining (EDM), at various locations along the sleeve/tube assembly.

Inspection results of the EDM notch samples with the sleeve probe revealed that the probability of detection increases with either increasing flaw length (constant flaw depth) or increasing flaw depth (constant flaw length). The EDM notch sample inspections showed a reliable EC detection threshold of both axial and circumferential notches of approximately 40 percent TW on the ID of the parent tube. Analysis (References 35 and 37) has shown the tube ID to be the location with the highest residual tensile stress in a hydraulically expanded tube. For inspection of the OD of the parent tube, the results of the EDM notched samples showed a reliable detection threshold of both axial and circumferential notches of approximately
[[ ]].

Along with the EDM notch sleeve mockup samples, additional ¾ inch diameter mill-annealed Alloy 600 tubes with SCC were obtained from EPRI, to confirm the detection capability of the sleeve probe with flaws that were more representative of the stress corrosion cracks that might develop during operation. To preserve the EPRI tube samples for future use in other projects, the sleeves were placed inside the tubes but were not expanded, which resulted in a slightly larger airgap between the sleeve and tube than would normally exist in the central region of an installed sleeve. The inspection results from the EPRI supplied samples with SCC showed a reliable EC detection threshold of cracks similar to the value for the EDM notches on the OD of the parent tube. Since the EPRI samples crack depths were not determined by destructive analysis, crack depths were determined based on ultrasonic examination results. These results are reasonably consistent with inspection results that have been obtained with other hydraulically expanded leak-limiting sleeves (Reference 25).

In addition to the eddy current qualification work that was performed at the Framatome facility in Lynchburg, VA, the licensee analyzed two supplemental EC datasets from tubes with existing SCC in the PNP SGs. The goal of acquiring the additional field data was to obtain additional data confirming reliable flaw detection in the SGs using the newly qualified EC sleeve probe technique.

For the initial field dataset investigation, 61 known tube flaws in the PNP SGs parent tubing were analyzed with both the sleeve probe (optimized for inspecting the sleeve/tube assembly) and the standard 3-coil +Point™ probe (optimized for flaw detection in tubes without sleeves). The results of this comparison showed that the sleeve probe readily detected and measured the 61 tube SCC indications, thereby confirming the ability of the sleeve probe to reliably detect cracking in parent tubes without sleeves (Reference 38). This flaw detection capability of the sleeve probe for unsleeved portions of SG tubing is important since the sleeve probe will also be used to inspect the unsleeved regions of a parent tube between adjacent sleeves.

The second supplemental EC dataset included 26 PNP SG tubes with ODSCC at the 5th hot leg lattice tube support location (Reference 38). The EC data from before and after sleeve installation were evaluated. The objectives of these inspections were: (1) to perform baseline inspection of the sleeve/tube assembly for comparison with future inspection results, and (2) to better understand detection thresholds for flaws in the parent tubing behind the sleeve. This dataset compares the EC signals obtained with a standard +Point™ probe (before sleeving) with the EC signals from the sleeve probe after installation of the H8 sleeves. The probability of detection curve from these inspections showed a high probability of detection of parent tube defects for flaw depths greater than approximately **[[** **]]**, after installation of an H8 sleeve. The staff notes that indication amplitudes behind sleeves were substantially reduced from the pre-sleeving indication with the standard 3-coil +Point probe but were still detectible with the sleeve probe, once the threshold detection depth was exceeded.

Based on the information provided in the LAR, the NRC staff finds the licensee's inspection qualification program acceptable since the licensee (a) will be inspecting the parent tube at the location where the sleeve joints will be established to ensure the region is acceptable for sleeving, (b) will be inspecting the sleeve/tube assembly and any parent tubing between sleeves at different tube support plate locations with the sleeve probe, and (c) has demonstrated that the sleeve probe is capable of reliably detecting structurally significant degradation in the sleeve/tube assembly, in the parent tube behind the sleeve, and in the parent tube between sleeves.

3.3.8 Future Sleeve Assembly Inspections

In a supplemental response to RAI #19, dated August 29, 2025, the licensee stated that all in service sleeve/tube assemblies will be inspected by the end of refueling outage 29 (1R29). The licensee also stated that inspection requirements at future outages beyond 1R29 will be established by the degradation assessment (DA) and operational assessment (OA) per TS 5.5.8.d and SR 3.4.17.1. The licensee also stated that future sampling inspections for in service sleeve/tube assemblies at refueling outages beyond 1R29 would include a minimum 50 percent sample size. If a flaw is identified in the sleeve/tube assembly during any sampling inspection, a scope expansion will be applied to the remaining 50 percent of the in service sleeve/tube assemblies per industry guidelines. The installation of tube sleeves increases the residual stress on the parent tube at the sleeve expansion locations. Framatome concluded that the increased stress from sleeve installation in the parent tube would be acceptable (Reference 24). The staff finds that the inspections that will be performed as part of the Steam Generator Program will detect any cracking, should it develop in this location, before it challenges tube integrity. Therefore, if the DA and OA allow for a future sampling inspection of sleeve/tube assemblies, it is important that this minimum sample scope achieves a high probability of detection, should SCC occur in the parent tubing of a sleeve assembly. A 50 percent inspection sample provides an approximately 94 percent probability of sampling at least one flawed location, if there are 4 assumed flaws in a population with over 16,000 inspection locations (2,000 sleeves with 8 expansions per sleeve). Detection of at least one flawed location would result in an inspection scope expansion. A high probability of flaw detection is an important component of maintaining tube integrity in the SG tubing. Therefore, the NRC staff finds that a license condition is necessary to ensure that a 100 percent inspection of in-service sleeve/tube assemblies will be performed by the end of 1R29 and a minimum 50 percent of the in-service sleeve/tube assemblies will be inspected by the end of each refueling outage after 1R29. The license condition to perform a 50 percent sleeve/tube assembly sampling inspection by the end of each refueling outage after 1R29 sets a minimum inspection scope only. The steam generator DA and OA may require up to 100 percent inspections in any given outage.

3.3.9 Sleeve Plugging Criteria

An evaluation was performed to the requirements outlined by the leakage and structural integrity performance criteria (SIPC), as defined in NEI 97-06 SG guidelines. Satisfying these requirements also effectively satisfies the original criteria for assessing degraded tubing defined in RG 1.121. The degraded tube analysis establishes a baseline for allowable tube degradation. However, as required by NEI 97-06, subsequent evaluations must be performed each inspection outage to address actual “as-found” degradation and actual plant-specific growth rates.

The structural evaluation of degraded tubes was performed using the criteria outlined/defined in the SIPC contained in Section 2.1 of NEI 97-06. The criteria were used with the tube loads from the design specification along with the appropriate flaw evaluation methods from the EPRI Flaw Handbook (Reference 39) to establish the tube structural limits. The accident leakage integrity of degraded tubes is assessed using the criteria defined in the performance criteria contained in Section 2.2 of NEI 97-06. Palisades is subject to a more restrictive criterion based on the plant's TSs. Furthermore, any leakage associated with the C* (C-star) alternate repair criteria is included in the accident leakage evaluation. C-star refers to repair of the SG, which involves plugging tubes with defects deep within the tubesheet as per TSs 5.5.8c.1 and c.2, whereas “tube repair” refers to sleeving tubes with tube defects in the lattice support area.

The results of this evaluation show that the performance criteria are satisfied, and they allow for non-destructive examination uncertainty and flaw growth. The material strength of the Alloy 690 sleeves and the applied loadings during normal operating and design-basis accident conditions would allow the Palisades technical specification tube plugging limit of greater than 40 percent TW to be applied to the H8 sleeves. However, given the absence of corrosion degradation during operation for Alloy 690 tubing, and because wear or other service-related degradation of the sleeve is not expected, the licensee will be using a plug-on-detection methodology for any sleeved tube that is found with service-induced degradation in the sleeve. The PNP SG Program currently requires plugging of crack-like indications on detection in the parent tubing, except in areas within the tubesheet covered by the existing C* alternate repair criteria. This same plug on detection criteria applies to the parent tubing portion of the pressure boundary in the sleeve joint regions. The SG tube SRs continue to ensure that defective tubes are repaired or removed from service upon detection.

3.3.10 Technical Conclusion

The tube support sleeves meet all pertinent design requirements with margins. The licensee analytical evaluations documented that ASME Section III stress analysis, seismic evaluation, flow induced vibration, fatigue usage, fatigue loadings, RG 1.121 criteria and thermal hydraulic impacts are acceptable. Mechanical tests were performed on sleeved mockup tubes to determine the leak rate properties. The results from the mechanical testing demonstrate that the sleeve assembly provides an adequate safety factor for normal operating and design-basis accident conditions. Mechanical testing determined the sleeve assembly would withstand cyclic loading resulting from power operation and other plant transients. The licensee also qualified an EC technique for sleeve assembly inspection and demonstrated that this technique was capable of detecting SCC if it occurs in the sleeve assembly and in parent tubing behind the sleeve after the sleeves are in service.

Based on the above, the NRC staff finds that the licensee has demonstrated the acceptability of the leak-limiting Alloy 690 sleeve repair in accordance with Appendix B to 10 CFR Part 50, GDCs 14, 15, 19, 30, 31, and 32 of Appendix A to 10 CFR Part 50, RG 1.121, and the applicable sections of ASME Code. As noted above, based on qualification testing duration, the leak-limiting Alloy 690 sleeve use approval is limited to 10 calendar years of operation after installation. In addition, sleeve installation approval is limited to the hot leg portion of the steam generator. Installation of sleeves in both the hot leg and cold leg support locations of the same tube would impede the ability to efficiently inspect the un-sleeved portion of the tube.

3.4 Evaluation of Proposed Changes to the TS

The licensee's proposed changes to the PNP TS are described in the February 11, 2025, LAR, in Enclosure 1, Section 2.2, "Description of Proposed Change." The changes are shown in the LAR Enclosure 2, "Technical Specification Page Markups," and Enclosure 3, "Retyped Technical Specification Pages." The changes proposed in this LAR (dated February 11, 2025), revise the affected TS and TS Bases included in the LAR dated December 14, 2023.

The licensee also submitted supplemental information dated May 29, 2025, to provide further clarity and corrections to the initial LAR TS documentation. For example, the supplemental information (not all-inclusive) proposed the following TS changes:

- TS 5.5.8 item c was revised by the licensee to address the plugging criteria for a sleeve. Specifically, sleeved tubes found by in-service inspection to contain a flaw in a sleeve or a flaw in the sleeve-to-tube joint shall be plugged.

- TS 5.5.8 item c was revised by the licensee to clarify that use of the SG alternate repair criteria, in lieu of the standard 40 percent depth-based criteria, is not a requirement, but an option, given that the tube could be plugged. As a result, the licensee proposed a change from "shall" to "may" in the last sentence of TS 5.5.8 item c. This change is consistent with the STS guidance.

The NRC staff finds that these changes clarified the initial LAR TS documentation and are consistent with the technical information and evaluation provided in Sections 3.1 through 3.3 of this SE.

As discussed in Section 1.2 of this SE, the licensee proposed revising TSs 3.4.1, 3.4.17, 5.5.8, and 5.6.8 to allow for the option to repair degraded SG tubes, define the acceptable method of repair, and address reporting requirements. As discussed in Sections 3.1 through 3.3 of this SE, the NRC staff reviewed the licensee's proposed use of leak-limiting Alloy 690 sleeves to repair SG tubes with degradation at the hot leg support locations and found it to be an acceptable alternative to plugging those tubes.

Including the option to repair SG tubes changes a TS LCO and associated remedial actions, TS SRs, and TS administrative controls. The NRC staff's findings on the TS changes are as follows:

- The NRC staff evaluated the proposed changes to TS 3.4.17 and determined that 10 CFR 50.36(c)(2) will continue to be met because 1) LCO 3.4.17, which was modified to address the option for repairing of SG tubes, specifies the lowest functional capability or performance levels of equipment required for safe operation of the facility; 2) when LCO 3.4.17 is not met, the licensee shall follow any remedial actions permitted by the technical specification until the condition can be met; and (3) the changes are consistent with applicable STS guidance.

- The NRC staff evaluated the proposed changes to TS 3.4.1 and TS 3.4.17 and determined that 10 CFR 50.36(c)(3) will continue to be met because the surveillance requirements for SR 3.4.1.3 and SR 3.4.17.2, which were modified to address the option for repairing of SG tubes consistent with applicable STS guidance, assure that the necessary quality of systems and components is maintained, that the facility operation will be within safety limits, and that the limiting conditions for operation will be met.

- The NRC staff evaluated the proposed changes to TS 5.5.8 and TS 5.6.8 and determined that 10 CFR 50.36(c)(5) will continue to be met because TS 5.5.8 and TS 5.6.8 include new administrative controls that address the option for repairing SG tubes (i.e., added SG tube repair criteria, SG tube repair methods, and reporting requirements for repaired tubes consistent with applicable STS guidance) necessary to assure operation of the facility in a safe manner.

4.0 DISPOSTION OF PUBLIC COMMENTS

On April 15, 2025 (90 FR 15722), the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating License, Opportunity to Comment, Request a Hearing, and Petition for Leave to Intervene" in the *Federal Register* associated with the proposed amendment request. In accordance with the requirements in 10 CFR 50.91, "Notice for public comment; State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. One public comment was received regarding the proposed amendment.

The NRC staff reviewed this comment and determined that the comment pertains to the SG not having been placed in wet layup when the plant was shutdown, and the resulting potential for future cracking or failure. Because the issue discussed in the public comment was quite general in nature and did not specifically address the basis for the proposed NSHC determination, this comment is not further addressed in this SE. However, the staff notes that this comment has been raised through other NRC processes related to PNP in a request for hearing, and is further being considered by the staff pursuant to 10 CFR 2.206, "Requests for action under this subpart." For example, the petitioners to intervene had raised similar issues in a hearing request filed on this LAR. The Atomic Safety and Licensing Board denied the hearing request on August 5, 2025 (ML25217A522). The 10 CFR 2.206 process is still in progress and the outcome will be documented in accordance with that regulatory process.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC staff's proposed no significant hazards consideration determination was published in the *Federal Register* on April 15, 2025 (90 FR 15722). On June 16, 2025, the NRC received an initial hearing request on this LAR from Beyond Nuclear, Don't Waste Michigan, Michigan Safe Energy Future, Three Mile Island Alert, and Nuclear Energy Information Service (collectively, Petitioners). On August 5, 2025 (ML25217A522), the Atomic Safety and Licensing Board (the Board) issued a Memorandum and Order denying their hearing request. On September 2, 2025 (ML25245A244), the Petitioners appealed the Board's decision; that appeal is pending before the Commission.

Under the Atomic Energy Act of 1954, as amended, and the NRC's regulations, the NRC staff may issue and make an amendment immediately effective, notwithstanding the pendency before the Commission of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has made a final determination that no significant hazards consideration is involved. The regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), Holtec provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

   Response: No.

   The Alloy 690 mechanical tube support plate (TSP) leak-limiting repair sleeves are designed using the applicable American Society of Mechanical Engineers (ASME)

Boiler and Pressure Vessel (B&PV) Code; therefore, they meet the design objectives of the original SG tubing. The applied stresses and fatigue usage for the repair sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of the repair sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. The acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide (RG) 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes. During the main steam line break (MSLB) leak testing performed as part of qualification, very small levels of primary-to-secondary leakage were measured and therefore no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The Alloy 690 repair sleeve depth-based structural limit is determined using the RG 1.121 guidance, Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines, and the pressure stress equation of ASME Code, Section III with additional margin added to account for configuration of long axial cracks at egg crate tube support plates. A bounding detection threshold value has been conservatively identified and statistically established to account for growth and determine the repair sleeve/tube assembly plugging limit. A sleeved tube is plugged upon detection of a degradation found in the sleeve/tube assembly.

Evaluation of the repaired SG tube testing and analysis indicates no detrimental effects on the sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at PNP. Corrosion testing and historical performance of sleeve/tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the way it is operated. The consequences of a hypothetical failure of the sleeve/tube assembly are bounded by the current SG tube rupture (SGTR) analysis described in the PNP UFSAR Revision 35. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the SGTR analysis and therefore, would result in lower total primary fluid mass release to the secondary system. A MSLB will not cause a SGTR because the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the PNP safety analysis. The minimal leakage that could occur during plant operation from repair of the sleeve/tube assembly is well within the plant leakage limits.

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

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

   Response: No.

   The Alloy 690 TSP leak-limiting repair sleeves are designed using the applicable ASME Code as guidance; therefore, it meets the objectives of the original SG tubing.

As a result, the functions of the SG will not be significantly affected by the installation of the proposed sleeves. The proposed repair sleeves do not interact with any other plant systems. Any accident because of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR accident analysis. The continued integrity of the installed sleeve/tube assembly is periodically verified by the inspection requirements in the proposed power operations technical specifications (POTS), as amended, and the requirement to plug sleeved tubes upon detection of a degradation. Implementation of the proposed amendment will have no significant effect on either the configuration of the plant, or the way it is operated.

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

3. <u>Does the proposed amendment involve a significant reduction in a margin of safety?</u>

Response: No.

The repair of degraded SG tubes with Alloy 690 TSP leak-limiting repair sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions and thereby maintains current core cooling margin as opposed to plugging the tube and taking it out of service. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Code used in the original SG design. The portions of the installed sleeve/tube assembly that represent the reactor coolant pressure boundary can be monitored for the initiation of sleeve/tube wall degradation and the affected tube can be plugged on upon detection of a degradation thereby restoring the integrity of the pressure boundary. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not different from the original SG tubes.

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

The NRC staff reviewed Holtec's no significant hazards consideration determination. Based on this review, the staff's evaluation of the underlying LAR as discussed above, and consideration of the public comments discussed in Section 4.0 of this SE, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued in accordance with the criteria in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment on September 15, 2025. The Michigan State official had no comments.

7.0 LICENSE CONDITION

In a supplement dated August 29, 2025, Palisades Energy provided commitments which the NRC staff determined should be restated as license conditions to support the NRC staff's safety determination. The staff's evaluation of these license conditions are documented in Section 3.3.8 of this SE.

License Conditions:

- Palisades Energy will inspect 100 percent of the steam generator in-service sleeve/tube assemblies by the end of cycle 29 refueling outage (1R29).

- Palisades Energy will perform an inspection of a minimum of 50 percent of all steam generator in-service sleeve/tube assemblies by the end of each refueling outage after 1R29. In the initial 50 percent sample, if a flaw is identified in the four expansion areas at either end of the sleeve of an in-service sleeve/tube assembly (A600 parent tube or A690 sleeve) or in the A690 sleeve between the expansion areas at the ends of the sleeve, a scope expansion will be applied to the remaining 50 percent of the in-service tube/sleeve assemblies. Identifying a flaw in the non-pressure boundary portion of the A600 parent tubing (behind the center portion of the sleeve) will not require a scope expansion. The Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines (ID 3002020909) will be used to select the minimum expansion scope.

8.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazards consideration, published in the *Federal Register* on April 15, 2025 (90 FR 15722) and there was one public comment on such findings, discussed in Section 4.0 of this SE. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment.

9.0 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.

10. REFERENCES

1. Palisades Nuclear Plant, License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated February 11, 2025. (ML25043A348).
2. Palisades Nuclear Plant - Response to Request for Additional Information regarding License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated May 29, 2025. (ML25149A013).
3. Palisades Nuclear Plant, Response to Second Request for Additional Information Regarding License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated July 30, 2025. (ML25211A324).
4. Holtec Palisades - Supplement to Response to Second Request for Additional Information Regarding License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated August 29, 2025. (ML25241A042).
5. Palisades Nuclear Plant – Second Supplement to Response to Second Request for Additional Information Regarding License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated September 23, 2025. (ML25266A097).
6. Palisades Nuclear Plant, Third Supplement to Response to Second Request for Additional Information Regarding License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving dated October 7, 2025, (ML25280A083)
7. Palisades Nuclear Plant, License Amendment Request to Revise Renewed Facility Operating License and Permanently Defueled Technical Specifications to Support Resumption of Power Operations dated December 14, 2023 (ML23348A148)
8. Palisades Nuclear Plant - Issuance of Amendment No. 276 Re: Amendment Request to Revise Renewed Facility Operating License and Permanently Defueled Technical Specifications to Support Resumption of Power Operations (EPID L-2023-LLA-0174) dated July 24, 2025 (ML25157A127).
9. Palisades Nuclear Plant - Certification of Permanent Cessation of Power Operations dated January 4, 2017 (ML17004A062).
10. Palisades Nuclear Plant - Certification of Permanent Cessation of Power Operations dated September 28, 2017 (ML17271A233).
11. Palisades Nuclear Plant - Supplement to Certification of Permanent Cessation of Power Operations dated October 19, 2017 (ML17292A032).
12. Palisades Nuclear Plant and Big Rock Point Application for Order Consenting to Transfers of Control of Licenses and Approving Conforming License Amendments dated December 23, 2020 (ML20358A075).
13. Palisades Nuclear Plant and Big Rock Point Plant - Order Approving Transfer of Licenses and Draft Conforming Administrative License Amendments (EPID L-2020-LLM-0003) (Letter) dated December 13, 2021 (ML21292A145).
14. Palisades Nuclear Plant - Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel dated June 13, 2022 (ML22164A067).
15. Palisades Nuclear Plant and Big Rock Point Plant - Issuance of Amendment Nos. 129 and 273 Re: Order Approving Transfer of Licenses and Conforming License

Amendments (EPIDs L-2022-LLM-0002 and L-2020-LLM-0003) dated June 28, 2022 (ML22173A173).

16. Palisades Nuclear Plant, Regulatory Path to Reauthorize Power Operations dated March 13, 2023 (ML23072A404).
17. Palisades Nuclear Plant - 2020 Steam Generator Tube Inspection Report dated March 25, 2021 (ML21084A077).
18. Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections", dated October 29, 2004 (ML043100526).
19. Palisades Nuclear Plant - Summary of Conference Call Regarding Steam Generator Tube Inspections (EPID L-2024-NFO-0008) dated October 1, 2024 (ML24267A296).
20. Regulatory Guide 1.121 Bases for Plugging Degraded PWR Steam Generator Tubes dated August 31, 1976 (ML003739366).
21. NUREG-0800, "Standard Review Plan, Section 16.0 Technical Specifications," dated March 9, 2010 (ML100351425).
22. Safety Evaluation by The Office of Nuclear Reactor Regulation Related to Amendment No. 189 To Facility Operating License DPR-20 Palisades Plant Consumers Energy Company Docket No. 50-255 dated November 30, 1999 (ML993510369).
23. NUREG-1432, Rev. 5, "Standard Technical Specifications Combustion Engineering Plants: Specifications" Vol. 1 and 2 dated September 30, 2021 (ML21258A421 and ML21258A424, respectively).
24. Framatome Document Number 51-9388710-001, "Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant," March 2024. (ML25043A348).
25. Beaver Valley, Unit 2, Issuance of Amendment RE: The use of Westinghouse Leak-Limiting Alloy 800 Sleeves for Steam Generator Tubes Repair dated September 30, 2009 (ML092590189).
26. Beaver Valley Power Station, Unit 2, 180-Day Steam Generator Tube Inspection Report dated May 1, 2025 (ML25121A190).
27. Palisades Nuclear Plant–Audit Summary Report in Support of Review of License Amendment Request Regarding Steam Generator Repair by Sleeving (EPID L-2025-LLA-0036) dated October 21, 2025 (ML25260A635).
28. *Steam Generator Program Guidelines-Revision 3*. Nuclear Energy Institute, Washington D.C.: 2011. NEI 97-06.
29. Framatome Document 08-9384357-001, Design Specification - "Hydraulically Expanded Sleeves for ¾ Inch Steam Generator Tubes at Palisades Nuclear Power Plant," January 2025. (Proprietary).
30. Framatome Document 32-9386352-000, Calculation Summary Sheet, "¾ Mechanical TSP Sleeve Leak Rate Analysis for Palisades," January 2025. (Proprietary).
31. Framatome Document 32-9387171-000, "Palisades SG Sleeve Primary Stress and Fatigue Evaluation," January 2025. (Proprietary).
32. Framatome Document 43-10202P-00, "Mechanical TSP Sleeve Qualification for ¾ Inch OD Tubes," December 1994. (Proprietary).
33. Framatome Document 51-9391941-000, "Palisades Eddy Current Sleeve Site Qualification," May 2025. (Proprietary).
34. Framatome Document 51-9321652-000, "Palisades Steam Generator Condition Monitoring for 1 R27 and Final Operational Assessment for Cycle 28," January 2021. (Proprietary).
35. Framatome Document 51-9385821-001, "Tube Support Plate (TSP) Sleeving Procedure Specification (SPS) for Palisades Steam Generator ¾" Tubes," May 2025. (Proprietary).

36. Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 8 dated June 30, 2016 (non-public).
37. W.B. Middlebrooks, D.L. Harrod and R.E. Gold, "Residual Stresses Associated with Hydraulic Expansion of Steam Generator Tubing into Tubesheets," Nuclear Engineering and Design 143, 1993.
38. Framatome Document, 51-9397841-000, "Consolidated White Paper: +Point Probe Performance Evaluation – Palisades Sleeve Project" dated September 12, 2025 (Proprietary).
39. *Steam Generator Management Program: Steam Generator Degradation Specific Management Flaw Handbook, Revision 2.* EPRI, Palo Alto, CA: 2015. 3002005426.
40. Comment (1) of Bruce Davis on Holtec Decommissioning International; Palisades Nuclear Plant License Amendment Request dated April 18, 2025 (ML25114A181)

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SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT NO. 281 REGARDING A CHANGE TO TECHNICAL SPECIFICATIONS TO ALLOW REPAIRING STEAM GENERATOR TUBES BY SLEEVING (EPID L-2025-LLA-0036) DATED OCTOBER 30, 2025

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