HOLTEC
PALISADES

27780 Blue Star Highway, Covert, MI 49043

PNP 2025-003

10 CFR 50.90

February 11, 2025

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

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

Subject: License Amendment Request to Revise Selected Permanently Defueled Technical Specifications to Support Repairing of Steam Generator Tubes by Sleeving

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, *Application for amendment of license, construction permit of early site permit*, Holtec[1] hereby requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) Renewed Facility Operating License (RFOL) DPR-20. The LAR would revise the Permanently Defueled Technical Specifications (PDTS) 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 LAR is contingent upon the prior approval of the LAR, dated December 14, 2023 (Reference 1), to reflect the resumption of power operations at PNP, which is currently under NRC review.

Specifically, this LAR proposes to revise the following PNP Technical Specifications (TSs) as provided in the Reference 1 LAR:

- TS 3.4.1, *Primary Coolant System (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 repairing or plugging SG tubes results in a specified flow reduction.

- The option to repair SG tubes is added to TS Limiting Condition for Operation (LCO) 3.4.17, *Steam Generator (SG) Tube Integrity*; and Administrative Controls TS 5.5.8, *Steam Generator (SG) Program*, which currently allow only tube plugging.

- Administrative Controls TS 5.6.8, *Steam Generator Tube Inspection Report*, is revised to add reporting requirements for repaired tubes.

[1] Holtec Palisades, LLC ("Holtec Palisades") is the licensed owner of PNP. Holtec Decommissioning International, LLC ("HDI") is the licensed operator of PNP while the facility is in decommissioning. Pursuant to the license transfer application submitted in connection with the PNP restart (Reference 5), licensed authority will transfer from HDI to Palisades Energy, LLC ("Palisades Energy") upon NRC's approval of the transition from decommissioning back to power operations. Holtec Palisades will remain the licensed owner of PNP.

By letter dated June 13, 2022 (Reference 2), Entergy Nuclear Operations, Inc. notified the NRC under 10 CFR 50.82, Termination of license, that it had permanently ceased operations and permanently removed fuel from the reactor vessel at PNP. Upon docketing the 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. However, shortly after PNP transitioned to a decommissioning facility, Holtec Palisades assumed ownership of PNP (Reference 3) and given the support from the Governor of the State of Michigan, Holtec commenced a project to return PNP to a power operations plant. The regulatory path to reauthorize power operations at PNP is described in HDI letter dated March 13, 2023 (Reference 4). In order to operate the PNP reactor, HDI also submitted a LAR to revise PNP RFOL DPR-20. The LAR would revise RFOL, Appendix A, Permanently Defueled Technical Specifications (PDTS), to reflect the resumption of power operations at PNP (Reference 1). The changes proposed in this LAR revise the affected TS and TS Bases pages (see Enclosures 2 through 4) included in the Reference 1 LAR. Accordingly, the TS changes contained in both LARs must be approved and implemented to support the resumption of power operations at PNP.

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

The proposed changes to the PNP PDTS are in accordance with 10 CFR 50.36, *Technical specifications*, paragraph 10 CFR 50.36(c)(5).

Enclosure 1 to this letter provides a detailed description and evaluation of the proposed changes to the PNP PDTS. Enclosure 2 contains a mark-up of the TS pages included in the Reference 1 LAR. Enclosure 3 contains the retyped TS pages containing the proposed changes. Enclosure 4 provides the changes to the TS Bases for information only. Enclosure 5 provides Non-Proprietary Framatome Document Number 51-9388710-001, *Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant.* Enclosure 5a provides Proprietary Enclosure 5, Framatome Document Number 51-9385467-002, *Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant*. Enclosure 6 provides the Framatome Affidavit, Proprietary Information Notice.

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

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

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 11, 2025.

Respectfully,

Jean A. Fleming

Digitally signed by Jean A. Fleming
DN: cn=Jean A. Fleming, o=Holtec Decommissioning International, LLC, ou=Regulatory and Environmental Affairs, email=J.Fleming@Holtec.com
Date: 2025.02.11 21:07:32 -05'00'

Jean A. Fleming
Vice President of Licensing and Regulatory Affairs
Holtec International

References:

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

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

3. NRC letter to HDI, "Palisades Nuclear Plant and Big Rock Point Plant – Issuance of Amendment Nos. 129 and 273 re: Order Approving Transfer of Licenses and Conforming Administrative License Amendments (EPIDS L-2022-LLM-0002 AND L-2020-LLM-0003)," dated June 28, 2022 (ADAMS Accession No. ML22173A173)

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

5. HDI letter to NRC, “Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments,” dated December 6, 2023 (ADAMS Accession No. ML23340A161)

<u>Enclosures:</u>

Enclosure 1 – Description and Evaluation of the Proposed Changes

Enclosure 2 – Technical Specifications Page Markups

Enclosure 3 – Retyped Technical Specification Pages

Enclosure 4 – Technical Specifications Bases Mark-Ups (For information only)

Enclosure 5 – Framatome Document Number 51-9388710-001, "Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant" (Non-Proprietary) (Non-Proprietary version of Framatome Document Number 51-9385467-002)

Enclosure 5a – Framatome Document Number 51-9385467-002, "Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant" (Proprietary) ***[Withhold Under 10 CFR 2.390(a)(4)]***

Enclosure 6 – Framatome Affidavit, Proprietary Information Notice

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

Enclosure 1

PNP 2025-003

Description and Evaluation of the Proposed Changes

TABLE OF CONTENTS

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGES

1.0 SUMMARY DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, *Application for amendment of license, construction permit of early site permit*, Holtec[1] hereby requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise the Palisades Nuclear Plant (PNP) Renewed Facility Operating License (RFOL) DPR-20. The LAR would revise the Permanently Defueled Technical Specifications (PDTS) 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 LAR is contingent upon the prior approval of the LAR, dated December 14, 2023 (Reference 1), to reflect the resumption of power operations at PNP, which is currently under NRC review.

The details regarding the design analyses, mechanical testing, corrosion resistance evaluation, nondestructive examination, installation process, and As Low As Reasonably Achievable (ALARA) aspects of the sleeve design for repair of the defective SG tubes at PNP are described in Non-Proprietary Framatome Document Number 51-9388710-001, *Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant*, which is provided as Enclosure 5. Enclosure 5a provides the Proprietary version of Enclosure 5, Framatome Document Number 51-9385467-002, *Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant*.

2.0 DETAILED DESCRIPTION

2.1 Reason for Proposed Change

By letter dated June 13, 2022 (Reference 2), Entergy Nuclear Operations, Inc. notified the NRC under 10 CFR 50.82, *Termination of license*, that it had permanently ceased operations and permanently removed fuel from the reactor vessel at PNP. Upon docketing the 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. However, shortly after PNP transitioned to a decommissioning facility, Holtec Palisades assumed ownership of PNP (Reference 3) and given the support from the Governor of the State of Michigan, Holtec commenced a project to return PNP to a power operations plant. The regulatory path to reauthorize power operations at PNP is described in HDI letter dated March 13, 2023 (Reference 4). In order to operate the PNP reactor, HDI also submitted a LAR to revise PNP

[1] Holtec Palisades, LLC ("Holtec Palisades") is the licensed owner of PNP. Holtec Decommissioning International, LLC ("HDI") is the licensed operator of PNP while the facility is in decommissioning. Pursuant to the license transfer application submitted in connection with the PNP restart (Reference 5), licensed authority will transfer from HDI to Palisades Energy, LLC ("Palisades Energy") upon NRC's approval of the transition from decommissioning back to power operations. Holtec Palisades will remain the licensed owner of PNP.

RFOL DPR-20. The LAR would revise RFOL, Appendix A, *Permanently Defueled Technical Specifications (PDTS)*, and Appendix B, *Environmental Protection Plan (EPP)*, to reflect the resumption of power operations at PNP (Reference 1). The changes proposed in this LAR revise the affected Technical Specification (TS) and TS Bases pages (see Enclosures 2 through 4 to this letter) included in the Reference 1 LAR. The approval of this LAR is required to make a mode change in support of resuming power operations. Accordingly, the TS changes contained in both LARs must be approved and implemented to support the resumption of full power operations at PNP.

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 5). At that meeting, Holtec presented a summary of the number of indications identified to date for each degradation mode (e.g., number of circumferential primary water stress corrosion cracking (PWSCC) indications at the expansion transition). For the most significant indications in each area, Holtec provided an estimate of the severity of the indication (e.g., voltage, depth, and length of the indication), including whether tube integrity (structural and accident induced leakage integrity) was maintained during the previous operating cycle. The plans for repairing or plugging of tubes were described.

The PNP TSs, as provided in the Reference1 LAR, allow defective tubes to be removed from service by installing plugs at both ends of the tube. The installation of SG tube plugs removes the heat transfer surface of the plugged tube from service and leads to a reduction in the primary coolant flow available for core cooling.

The proposed LAR revises the PNP TSs to permit the use of Alloy 690 repair sleeves. There are three distinct advantages associated with the Alloy 690 repair sleeves. First, no welding, brazing, or heat treatment is required during sleeve installation. Second, the strain within the tube is low, thereby reducing the likelihood of future degradation due to stress-influenced mechanisms. Third, the sleeve is fabricated from thermally treated Alloy 690 material, which provides enhanced corrosion resistance compared to the SG tubing. The proposed LAR would also prevent unnecessary plugging of SG tubes, if appropriate, prior to the restart of the unit and subsequent refueling outages.

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

2.2 Description of Proposed Change

The following is a detailed description of the PNP TS changes proposed by this LAR:

- 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 repairing or plugging SG tubes results in the same primary flow reduction as plugging 10 tubes.

- The option to repair SG tubes is being added to TS limiting condition for operation (LCO) 3.4.17, *Steam Generator (SG) Tube Integrity*. This TS LCO currently allows only tube plugging.

- Administrative Controls TS 5.5.8, *Steam Generator (SG) Program*, is being revised to allow the option to repair SG tubes, update the provisions for SG tube repair criteria, and specify the allowable SG tube repair methods.

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

Enclosure 2 to this letter provides the TS pages included in Reference 1 marked-up to show the proposed changes. Enclosure 3 provides the retyped pages. Enclosure 4 provides the TS Bases pages marked-up to show the proposed changes. The changes to the TS Bases are provided for information only. Enclosures 5 and 5a provide non-proprietary and proprietary copies of the supporting technical report.

The proposed changes to the PNP PDTS are in accordance with 10 CFR 50.36, *Technical Specifications*, Subsection (c)(5).

3.0 TECHNICAL EVALUATION

3.1 Introduction

Sleeving is a method used to repair defective SG tubes and thus keep the tubes in service. The sleeves being proposed for use at the PNP are made of Alloy 690. The primary criterion for the selection of the Alloy 690 sleeve material was its excellent corrosion resistance in both the primary and secondary side Pressurized Water Reactor (PWR) environments.

The sleeve qualification and testing comply with the in-service Inspection and Repair requirements of American Society of Mechanical Engineers (ASME) Code Section XI, IWA-4721, and more specifically Subsection IWA-4725 pertaining to the Expansion Requirements for sleeves, 2007 Edition with Addenda through 2008. The sleeve design calculations use allowable stresses based on the strength properties listed in ASME Boiler and Pressure Vessel (B&PV) Section II for Alloy 600 and Alloy 690.

The structural adequacy of the recirculating steam generator (RSG) sleeves was evaluated for pressure thickness, external pressure, seismic (Operating Basis Earthquake (OBE)), and flow induced vibration in accordance with the ASME B&PV Code. A stress and fatigue evaluation was performed consistent with Regulatory Guide (RG) 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes*, and Nuclear Energy Institute (NEI) 97-06, *Steam Generator Program Guidelines*, to establish the plugging criteria for the sleeve and for the tube in the expansion regions.

Mechanical tests were performed on sleeved mock-up tubes to determine the leak rate properties. These tests included initial leak rates, axial, thermal, and pressure cycling, and final leak rates at normal operating and main steam line break (MSLB) conditions.

3.2 System Description

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 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 (Reference 7), Table 4-4, *Steam Generator Parameters*. 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 Enclosure and UFSAR Section 4.3.4, *Steam Generator*.

3.3 Applicable Safety Analyses

The principal accident associated with this proposed change is the SG tube rupture (SGTR) event. The consequences associated with a SGTR event are discussed in UFSAR Section 14.15, *Steam Generator Tube Rupture With A Loss of Offsite Power*.

The SGTR accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for TS LCO 3.4.17. The analysis of a SGTR event assumes a primary to secondary leakage rate that bounds the operational leakage rate limits specified in TS LCO 3.4.13, *PCS Operational LEAKAGE*, plus the leakage rate associated with a double-ended rupture of a single tube.

The SGTR accident is a penetration of the barrier between the PCS and the main steam system which results from the failure of a SG U-tube. Integrity of the barrier between the PCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. Following a reactor trip and turbine trip, the radioactive fluid is released through the SG safety or atmospheric dump valves as a result of an assumed loss of normal AC power.

As discussed in UFSAR Section 14.15.4, the radiological releases calculated for the SGTR event with a loss of offsite power are well below the limits for offsite doses. The PCS and secondary system pressures during the SGTR remain below 110% of the design pressure limits, thus, assuring the integrity of these systems.

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 0.3 gpm per SG. The accident induced leakage rate includes any primary to secondary leakage existing prior to the accident in addition to primary to secondary leakage induced during the accident.

The safety analyses of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. TS SR 3.4.1.3 is being revised to include appropriate surveillance of PCS flow rate.

3.4 Sleeve Design

3.4.1 Description of Sleeves

Sleeving is a method used to repair defective SG tubes and thus keep the tubes in service. A lattice tube support plate (TSP) sleeve is designed to repair tube degradation occurring at TSP intersections of SG tubes. A mechanical sleeve is a tube segment that is inserted into an existing SG tube and expanded to create an interference fit between the sleeve and the tube. The proposed sleeve uses hydraulic expansion to create the interference fit.

The tubing material for the sleeve is thermally treated Alloy 690. Section 3.4.3 discusses the basis for selecting Alloy 690 as the sleeve material. The sleeve is fabricated with sealing ribs on the outside diameter (OD) of eight expansion areas. The profile of these ribs, when combined with the sleeve expansions, gives the sleeve improved leak limiting characteristics compared to the expansion of a smooth sleeve. The free span sleeve/tube joints are produced by a controlled hydraulic expansion process. A total of eight joints are produced simultaneously, with four above and four below the TSP.

Figure 3.4-1 illustrates a typical installation. A longitudinal section of the sleeve is shown in Figure 3.4-2.

For additional information, refer to Enclosures 5 and 5a, Section 5.0.

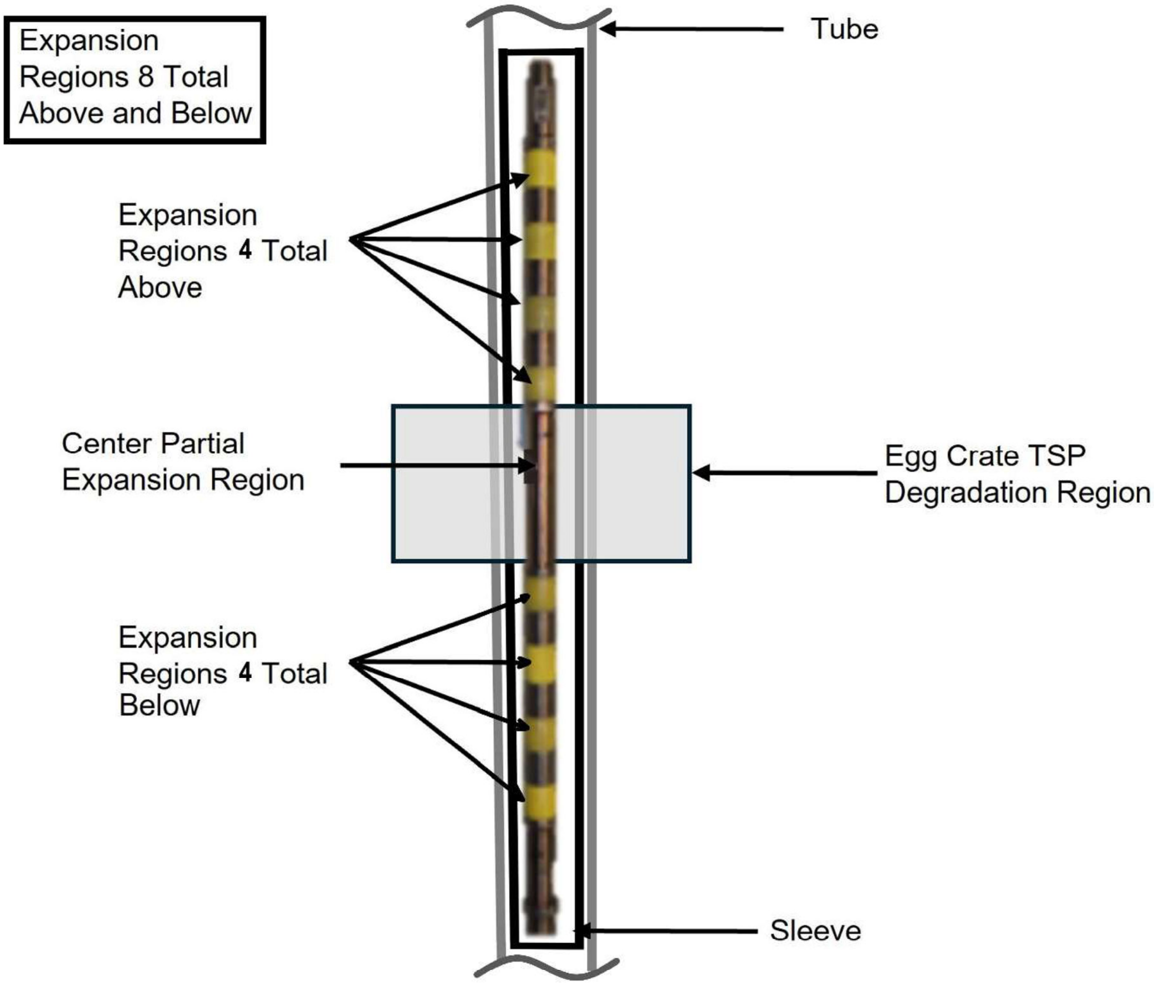


Figure 3.4-1
Typical Sleeve Installation

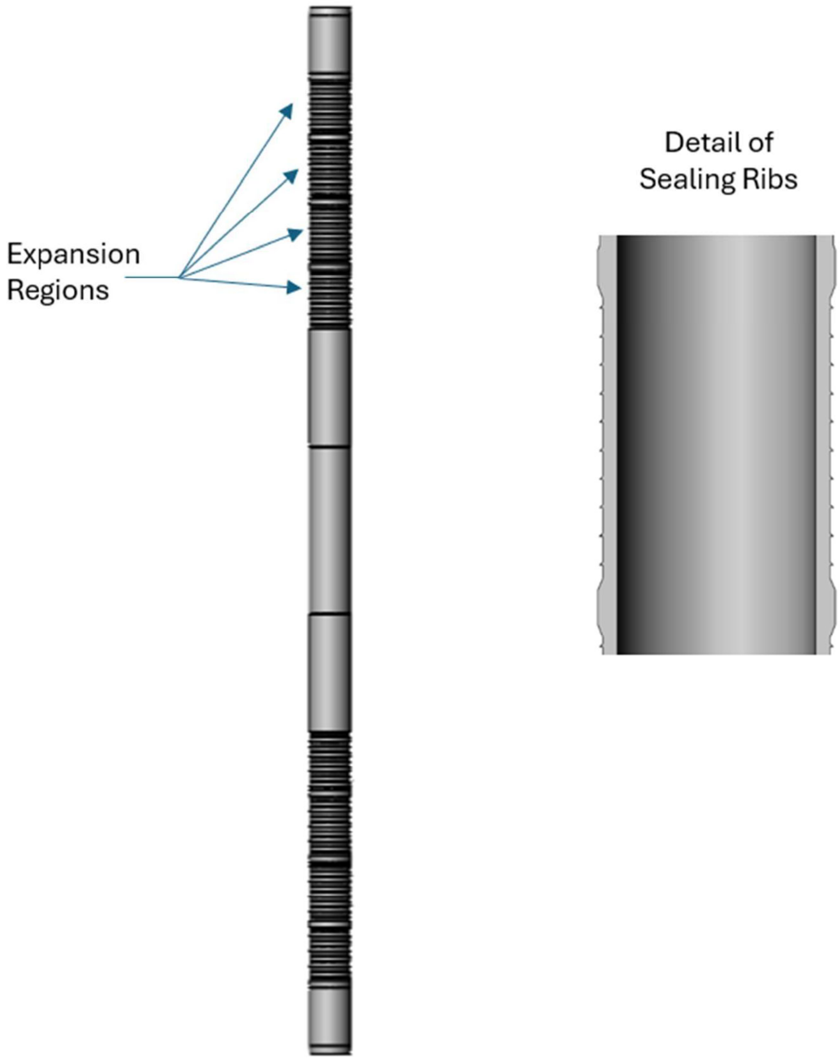


Figure 3.4-2
Sleeve Section Drawing

3.4.2 Sleeve Design Conditions

The sleeves are designed and constructed in accordance with ASME B&PV Code Section III, Subsection NB. The sleeve design was qualified by testing for leakage in accordance with ASME B&PV Code Section XI, IWA-4725.

The objective of installing sleeves in SG tubes is twofold. The sleeves must maintain structural integrity of the SG tube during normal operating and postulated accident conditions and the sleeve must limit the primary to secondary leakage in the event of a through wall defect in the section of the SG tube spanned by the sleeve.

Numerous tests and analyses were performed in the 1994 timeframe to demonstrate the capability of the mechanical TSP sleeve for Westinghouse Electric Company (WEC) SGs and a 10-year life. An applicability assessment was performed to demonstrate that the numerous tests and analyses bound the PNP SGs. The assessment demonstrated that PNP is bounded by several conservative design basis values from WEC plants.

To qualify the sleeves, a set of sleeve criteria has been established to demonstrate the acceptability of the Alloy 690 sleeve for PNP. The criteria and justification are provided in Table 3.4.2-1. The Section column provides a reference to the appropriate section in Enclosures 5 and 5a that provide the justification that the criteria have been met for PNP.

Table 3.4.2-1
Sleeve Criteria

Criterion	Approach	Results	Section
Exposure of the repair assembly to various primary and secondary chemistries without a loss of function.	Literature research and operating experience for Alloy 690 material.	Alloy 690 is extremely resistant to the chemical environments present in PWR primary and secondary water chemistries.	6
Repair sleeve-tube assembly structural integrity must be maintained for normal and accident conditions.	Repair sleeve-tube assembly meet applicable ASME Code requirements, including fatigue.	Framatome sleeve meets the requirements of ASME Code including Fatigue.	7
Sleeve/tube joint load capability 3 times normal Δp and 1.4 times steam line break Δp even for a severed tube. Does not collapse sleeve during LOCA.	RG 1.121 and NEI 97-06 plugging criteria analysis.	Sleeve plugging criteria established to meet 3 times normal Δp and 1.4 times steam line break Δp even for a severed tube.	7

Criterion	Approach	Results	Section
Sleeve/Tube Joint deflection capability is sufficient for thermal expansions effects.	No degradation of leak limiting or structural load capability for worst case thermal expansion cycles.	No sleeve motion or degradation was noted from cycle testing.	8
Pressurization of annulus between the sleeve and tube does not fail during operation or design basis accidents.	Prevention of sleeve failure based on pressure testing.	No sleeve motion or degradation as noted from cycle testing.	8
Repair sleeve installation to satisfactorily limit leakage in any direction and under normal and accident conditions.	Allowable leakage established by PNP TS.	Leak rate estimates per sleeve.	8
Repair sleeve effect on system flow rate and heat transfer capability of the SG is acceptable.	Allowable reduction in coolant flow rate is limited per PNP TS. Steam pressure reduction due to reduced heat transfer is limited for commercial considerations.	The sleeve effect on system flow rate and heat transfer capability of the SG was calculated.	7
Non-destructive examination of the tube and sleeve pressure boundary with levels of detectability sufficient to show structural adequacy.	Periodic examination of tubes and sleeves are required.	Eddy current inspection capability is demonstrated as capable of detecting degradation within a sleeved tube.	10

3.4.3 Alloy 690 Selection

The mechanical TSP sleeve material is thermally treated Alloy 690. The primary criterion for the selection of the Alloy 690 sleeve material was its excellent corrosion resistance in both the primary and secondary side Pressurized Water Reactor (PWR) environments. Alloy 690 is virtually impervious to PWSCC and has demonstrated greatly improved resistance to outside diameter stress corrosion cracking (ODSCC).

Alloy 690 and its weld metals have been used in most PWR replacement component items, such as SG tubing, repair sleeves, plugs, pressurizer heater sleeves, and vessel closure head nozzles. Alloy 690 has been in service since the late 1980’s and no cracking of the Alloy 690 materials has been observed in the U.S. or international plants. Alloy 690 has proven to be the best alloy currently available for both primary and secondary side corrosion concerns.

For additional information, refer to Enclosures 5 and 5a, Section 6.

3.5 Supporting Sleeve Analyses

3.5.1 ASME Section III Design Analysis

As described below, the analytical evaluations documented the required ASME Section III stress analysis, seismic, flow induced vibration (FIV), fatigue usage, fatigue loadings, RG 1.121 criteria, and thermal-hydraulic impacts.

a. <u>Sleeve Design Calculations</u>

The sleeve design calculations use allowable stresses based on the strength properties listed in ASME B&PV Section II for Alloy 600 and Alloy 690. The structural adequacy of the SG sleeves was evaluated for pressure thickness, external pressure, seismic (OBE), and flow induced vibration in accordance with the ASME B&PV Code. An applicability assessment was performed to assess the primary stress qualification and to recalculate the fatigue usage factor for the PNP specific transients and sleeve conditions. The allowable external pressures for the sleeves and tubes were calculated per paragraph NB-3133.3. The sleeve exceeds the strength of the original tube for external pressure loadings. The analysis of the hydraulic expansion region of the sleeve/tube assembly uses the criteria of Subsection NB of the ASME B&PV Code to establish allowable stresses.

The structural integrity of the freespan hydraulic expansion was evaluated for design basis loads using a finite element analysis (FEA). The expanded sleeve/tube geometry was subjected to a loading condition consisting of internal (primary side) and external (secondary side) pressures which correspond to the design differential pressure conditions. In addition, the stress analysis considered seismic (OBE) loads and flow induced vibration loads. The resulting primary stresses were evaluated per the design requirements of the ASME B&PV Code. The stresses in the tube and sleeve satisfy the primary stress limits. The secondary-to-primary pressure differential was not limiting.

The TSP sleeve meets the ASME Section III stress and fatigue usage limits and are acceptable for PNP. The TSP sleeve meets the design requirement of maintaining structural adequacy during normal operating and accident conditions.

For additional information, including details regarding the primary stress assessment, assessment of pressure loads, assessment of OBE loads, assessment of FIV loads, and fatigue analysis, refer to Enclosures 5 and 5a, Sections 7.1.1 through 7.1.5.

b. *<u>Fatigue Test Loadings</u>*

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 Section III, Appendix II, and are based on the number of test assemblies and various factors relating the test conditions to the actual operating conditions.

The test loads were originally calculated for the design conditions and transients of the WEC SGs. The fatigue cycling performed for the WEC SGs encompasses the fatigue loadings applicable to the mechanical TSP sleeves installed at PNP.

For additional information, refer to Enclosures 5 and 5a, Section 7.2.

c. *Sleeve Plugging Criteria*

The evaluation was performed to the requirements outlined in the leakage and structural integrity performance criteria (SIPC) as defined in NEI 97-06 SG guidelines. Satisfying these requirements also effectively satisfy 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 Electric Power Research Institute (EPRI) Flaw Handbook 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. PNP 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 TSP area in accordance with Enclosures 5 and 5a.

The operational leakage integrity of degraded tubes is assessed using the criteria outlined/defined in the performance criteria contained in Section 2.3 of NEI 97-06. PNP has implemented an administrative operational leakage limit of 0.05 gallons per minute (72 gallons per day).

The results of this evaluation show that the performance criteria are satisfied, and they allow for non-destructive examination (NDE) uncertainty and flaw growth. The PNP TS tube plugging limit of >40% through wall (TW) is also applicable to the TSP 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 alternate repair criteria. This same plug-on-detection criteria applies to the parent tubing portion of the pressure boundary in the sleeve joint regions. SG tube SRs continue to ensure that defective tubes are repaired or removed from service upon detection.

For additional information, including the analytical methodology, refer to Enclosures 5 and 5a, Section 7.3.

d. *Thermal-Hydraulic Analysis*

The effect of TSP sleeve installation on the SG performance was analyzed for heat transfer, flow restriction, and SG capacity. The sleeve to plug ratio will vary depending on the sleeve length, number of plugs installed, and the number of sleeves installed. The sleeve to plug ratio is a minimum value which increases with the number of sleeves installed

At the conclusion of each SG eddy current test (ECT) inspection, the effective plugged percentage of each SG is calculated and reported in the Operational Assessment for the next operating cycle. Comparison to the allowed plugging percentage will demonstrate proper operation of the SG with respect to heat transfer for the next cycle. In addition, SR 3.4.1.3 is revised to require the PCS total flow rate to be verified within limits whenever repairing or plugging SG tubes results in the same primary flow reduction as plugging 10 tubes.

For additional information, refer to Enclosures 5 and 5a, Section 7.4.

e. *Sleeved Tube Seismic Considerations*

PNP is in an area of very low seismic activity. The postulated OBE loads which were used are much greater than the PNP design basis. As discussed in UFSAR Section 5.7.1.1, ground accelerations for the operational basis earthquake used for containment design purposes and seismic Class I structures are 0.10g applied horizontally and 0.07g applied vertically. In addition, ground accelerations of 0.2g horizontal and 0.13g vertical are used for the design basis earthquake. The recommended surface acceleration value was 0.05g. A conservative value was used for the Facility design earthquake (OBE).

The effect of TSP sleeve installation on the stresses due to seismic loading, as well as other concurrent design basis loads in the tubes, was evaluated and shown to be acceptable.

For additional information, refer to Enclosures 5 and 5a, Section 7.5.

f. *Flow Included Vibration*

The FIV analyses evaluated fluid-elastic stability margins (FSM) and random turbulence vibration response for TSP sleeves installed at the TSPs on the hot leg. The natural frequencies and mode shapes of the SG tube and sleeve combination were determined. The FIV tube model included the hot leg, U-bend, and cold leg tubing from tubesheet to tubesheet. The FIV analyses indicated that the sleeves are acceptable for installation based on FSM and random turbulence vibration considerations.

For additional information, refer to Enclosures 5 and 5a, Section 7.6.

g. <u>Conclusion</u>

The TSP sleeves meet the pertinent design requirements with margins. The analytical evaluations documented the ASME Section III stress analysis, seismic, flow induced vibration, fatigue usage, fatigue loadings, RG 1.121 criteria, and thermal-hydraulic impacts and the results are acceptable.

The Alloy 690 leak limiting repair sleeves are designed using the applicable ASME B&PV Code; therefore, they meet the design objectives of the current SG tubing. The applied stresses and fatigue usage for the repaired sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of 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 RG 1.121.

The Alloy 690 repair sleeve depth-based structural limit was determined using the RG 1.121 guidance and the pressure stress equation of ASME Code, Section III. The proposed PNP Administrative Controls TS 5.5.8c limit of a depth equal to or exceeding 40% of the nominal tube wall thickness is conservative. A sleeved tube is plugged on detection of degradation in the sleeve/tube assembly.

Evaluation of the repaired SG tube by 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. The sleeve is fabricated from Alloy 690 which provides enhanced corrosion resistance compared to the current SG tubing.

The implementation of the proposed sleeve 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 SGTR analysis described in the PNP UFSAR, Section 14.15.

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 would result in lower total primary fluid mass release to the secondary system. The minimal leakage that could occur from the repaired sleeve/tube assembly during plant operation is well within the TS leakage limits.

3.5.2 ASME Section XI Mechanical Testing of the Sleeves

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.

a. Specimens

The mechanical test specimens are described in Enclosures 5 and 5a, Section 8.1. The sleeves were installed using the Framatome hydraulic expansion system. A range of tube 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 ribs on the OD of the sleeve to better engage the tube.

b. Mechanical Tests

Details of each type of mechanical test are provided below.

b.1 Primary Side Room Temperature Leak Test

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.

Refer to Enclosures 5 and 5a, Section 8.2.1 for additional information.

b.2 Operating Temperature Leak Test

Operating temperature leak testing was also performed on various specimens during the qualification process.

For additional information, refer to Enclosures 5 and 5a, Section 8.2.2.

c. Cyclic Testing

The adequacy of the sleeve and its joints to withstand operational loading was demonstrated by subjecting the 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.

c.1 Axial Cycling

Several specimens were axially cycled to represent 10 years of structural design life. None of the specimens recorded sleeve joint motion.

c.2 Thermal Cycling

Several specimens were thermally cycled. None of the samples recorded sleeve motion during this testing.

c.3 Pressure Cycling

Pressure cycling was performed on several specimens. This test represents the startup and shutdown pressure cycling to which a sleeve will be exposed during 10 years of design life. No sleeve-to-tube movement was noted during testing.

For additional information on the axial, thermal and pressure cycling tests, refer to Enclosures 5 and 5a, Section 8.2.3.

d. *Main Steam Line Break Leak Testing*

After the leak and cyclic tests were complete, MSLB leak testing was performed to represent the maximum pressure expected from an MSLB accident. The leak rates measured during this testing, performed at room temperature, were higher than those previously recorded during room temperature leak testing. However, no joint failure was noted during testing.

For additional information, refer to Enclosures 5 and 5a, Section 8.2.4.

e. *Analysis of Leakage Rates*

After qualification testing was complete on the mechanical TSP sleeve samples, data analysis was performed to determine the expected leak rates. Leak rate values were determined for both beginning and end of life leakage, with beginning of life defined as the leakage prior to cyclic loading and end of life after the 10-year cyclic loading. The operational leak rate is monitored through plant equipment and does not limit the total number of sleeves that can be installed. The MSLB leakage value does limit the total number of sleeves that can be installed. Based solely on leakage considerations, a large number of sleeves could be installed in each SG.

The PNP TS Operational and Accident-Induced leakage limits assure that any postulated leakage in the limiting SG will be less than the bounding faulted condition leakage necessary to ensure that offsite doses remain a small fraction of the 10 CFR Part 100, *Reactor Site Criteria,* values and that control room doses remain within 10 CFR 50 Appendix A, *General Design Criteria for Nuclear Power Plants,* General Design Criteria (GDC) 19, *Control room,* limits. The operational and accident-induced leakage contribution from the installed sleeves is held within the PNP TS operational and accident-Induced leakage limits.

For additional information, refer to Enclosures 5 and 5a, Section 8.2.5.

f. *ASME Section XI Testing Conclusions*

Mechanical tests were performed on sleeved mock-up tubes to determine the 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 of the mechanical testing of the sleeve samples demonstrates that they provide an adequate safety factor for normal operating and postulated accident conditions. The mechanical testing also determined that the installed sleeve would withstand the cyclical loading resulting from power changes and other transients.

These conclusions are provided in Enclosures 5 and 5a, Section 8.3.

3.6 Summary of Technical Analyses

The sleeve qualification and testing comply with the in-service Inspection and Repair requirements of ASME Code Section XI, IWA-4721, and more specifically Subsection IWA-4725 pertaining to the expansion requirements for sleeves, 2007 Edition with Addenda through 2008.

The sleeve qualification included axial load, pressure, and temperature cycling. The sleeves were axially cycled for 10 years' worth of plant transients and thus have been qualified beyond the PNP currently licensed plant life. Fatigue loadings for use in mechanical qualification testing of the sleeve and tube joint were calculated per the ASME B&PV Code. A stress and fatigue evaluation was performed consistent with RG 1.121 and NEI 97-06 to establish the plugging criteria for the sleeve and for the tube in the expansion regions.

3.7 Sleeve Installation

The sleeve proposed for installation in the PNP SG has been installed over 6,000 times in non-SG heat exchangers. Of which, almost 1,000 were in ASME Section III heat exchangers, including CCW and containment spray heat exchangers. The sleeves installed in the ASME Section III heat exchangers fall under 10 CFR Part 21, *Reporting of Defects and Noncompliance*, and the vendor has received no notifications regarding 10 CFR Part 21 pertaining to these sleeves.

The installation of the mechanical TSP sleeve is accomplished remotely by tooling attachments mounted on a manipulator. The sleeve installation and inspection methods used minimize the personnel radiation exposures in accordance with ALARA principles.

The pre-installation testing includes an ECT of the tubes to be sleeved to verify there is no degradation of the tube in the upper and lower expansions. If not already known, the distance from the TSP to the diagonal supports can also be confirmed by ECT to assure the sleeve can be installed in locations near the diagonal supports. Any observed dents/dings in the target tube are noted for further testing.

After the sleeves are installed, an eddy current examination is performed to verify the correct location of the sleeve relative to the defect region and to obtain a baseline inspection of the sleeve and tube. This allows future comparisons of the sleeve and tube conditions. The ability to detect flaws in the parent and/or sleeve tubing has previously been demonstrated by previous eddy current qualifications. If a sleeved SG tube is found to have an unacceptable defect in the pressure boundary portion of the tube or sleeve, the SG tube may be taken out of service by installing the standard, site approved, mechanical plugs at both ends of the tube.

For additional sleeve installation information, refer to Enclosures 5 and 5a, Sections 9.0 through 9.4. Discussions of ECT requirements for tubing with sleeves, eddy current qualification, and generic eddy current sleeving experience are provided in Enclosures 5 and 5a, Section 10.

3.8 ALARA Radiation Dose Considerations

The SG repair operation is designed to minimize personnel radiation exposure during the installation of the sleeves. To achieve this, the manipulator is installed from the manway without entering the SG and is operated remotely from a control station outside the containment building. The sleeve delivery system allows the sleeve to be positioned on the expansion tool from outside the SG. The expansion tool and sleeve are then delivered into the SG tube remotely.

Additional details are provided in Enclosures 5 and 5a, Section 9.5.

3.9 ASME Code Reconciliation

The design and qualification of the sleeve is governed by applicable industry codes and standards as summarized in Table 3.9-1. The ASME B&PV Code is the basic governing document for numerous aspects of the design, including determining test loads, performing structural analyses, procuring material, establishing the sleeve procedure qualification, and preparing the sleeving procedure specification. The mechanical TSP sleeve meets the requirements of the ASME B&PV Code.

The sleeves are designed and constructed in accordance with ASME B&PV Code Section III, Subsection NB. The sleeve design and sleeving procedure were qualified by testing for leakage in accordance with ASME B&PV Code Section XI, IWA-4725

The repair activities are completed in accordance with the NRC approved version of ASME Section XI, 2007 Edition with Addenda through 2008, which is the PNP approved Code of record. The sleeves 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 the Section XI, 1989 Edition used to qualify the sleeve was IWB-4300, *Heat Exchanger Tube Sleeving*. The relevant Section in the 2007 Edition with 2008 Addenda of the Code is IWA-4725, *Expansion*. A comparison of the Code Editions concluded that there are no changes to technical requirements. Therefore, the technical requirements of the ASME B&PV Code, Section XI, 2007 Edition with Addenda through 2008 are met.

The materials, design, fabrication, and examination of the PNP SG tubes meet the requirements of ASME B&PV Code, Section III, 1977 Edition. The materials, design, fabrication, and examination of the sleeves meet the requirements of ASME Section XI, 1989 Edition and the 2007 Edition with Addenda through 2008. Use of a later code is permitted by ASME Section XI, IWA-4221, provided reconciliation is performed in accordance with IWA-4222 through IWA-4226 as applicable. Those paragraphs require the sleeves be evaluated for changes to the weight, configuration, and pressure-temperature rating of the SGs. The sleeve installation, regarding the weight, configuration, and pressure-temperature ratings of the SGs, meets the requirements of IWA-4222 through IWA-4226.

Table 3.9-1
Applicable Codes and Standards

Application	Criteria
Structural Design of the Sleeve	ASME B&PV Code Section II & III
Sleeve Plugging Limit	NRC Reg. Guide 1.121 NEI 97-06
Material Procurement	ASME B&PV Code Section II & III
Mechanical Sleeve Qualification	ASME B&PV Code Section XI
Sleeve NDE	ASME B&PV Code Section V & XI

3.10 Detection of Potential Tube Failures

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

4.0 REGULATORY EVALUATION

The following identifies the applicable regulatory requirements and guidance documents and describes how this proposal is in conformance with those requirements and guidance documents.

4.1 10 CFR 50.36, Technical Specifications

In accordance with 10 CFR 50.36, TSs are required to include items in the following five categories: (1) Safety Limits, limiting safety system settings, and limiting control settings; (2) Limiting Conditions for Operation (LCOs); (3) Surveillance Requirements (SRs); (4) Design Features; and (5) Administrative Controls.

This LAR is proposing changes to the Administrative Controls TSs 5.5.8 and 5.6.8, with conforming changes to TS SR 3.4.1.3 and TS LCO 3.4.17, consistent with the proposed return to power operation status of the plant.

The PNP TSs and proposed changes satisfy 10 CFR 50.36.

4.2 10 CFR 50.36(c)(2)(ii)

Paragraph (c)(2)(ii) of 10 CFR 50.36 requires, in part, that a TS LCO be established for each item meeting one or more of the following criteria:

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

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

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

PNP TS LCOs 3.4.1, *PCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits*, and 3.4.17, *Steam Generator (SG) Tube Integrity*, satisfy these requirements. TS LCO 3.4.1 provides requirements for maintaining PCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The PCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). SG tubes are an integral part of the PCPB and, as such, are relied on to maintain the primary system’s pressure and inventory. SG tube integrity (TS LCO 3.4.17) satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

4.3 10 CFR 50.36(c)(3)

Paragraph (c)(3) of 10 CFR 50.36 states, in part, that TS SRs 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 LCO will be met.

The TS SRs provided for LCOs 3.4.1 and 3.4.17, as modified by this LAR, satisfy this requirement.

4.4 10 CFR 50.36(c)(5)

Paragraph (c)(5) of 10 CFR 50.36 states, in part, that TS Administrative Controls are provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

PNP Administrative Controls TSs 5.5.8 and 5.6.8, as modified by this LAR, contain inspection and reporting provisions to ensure SG tube integrity is maintained, which satisfy this requirement.

4.5 10 CFR 50, Appendix A, General Design Criteria

Appendix A, *General Design Criteria for Nuclear Power Plants*, to 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities*, establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The following GDCs are applicable to, and appropriate for this review. The following discussions specifically refer to the GDCs and their interpretations which existed on July 7, 1971. These discussions should not be construed as commitments to comply with any interpretation document issued after this date. Any commitments to later design requirements are documented separately in the UFSAR:

- GDC 14, *Reactor Coolant Pressure Boundary*, which requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

  Conformance: The reactor coolant pressure boundary meets Criterion 14 as discussed in UFSAR Section 5.1.3.5. The sleeve replaces the pressure boundary at the defect location and precludes a potential tube rupture. The qualification and testing for the sleeve considered operating and faulted conditions at temperature for a tube defect to sever during continued operation. If the tube were to sever at the defect location, it is possible that small leakage past the sleeve joints may occur but are shown to be very small. Many sleeves could be installed in tubes and still be well within the plant TS leakage limits, which preserves the Reactor Coolant System (RCS) pressure boundary. The installed sleeve enhances tube integrity and precludes tube rupture for the installed location(s).

  Therefore, the reactor coolant pressure boundary, containment boundary, and tube-bundle integrity will not be adversely affected by the application of qualified Alloy 690 sleeves.

- GDC 15, *Reactor Coolant System Design*, which requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

  Conformance: The RCS, including its auxiliary, control and protection systems, was designed in compliance with Criterion 15 as discussed in UFSAR Sections 5.1.3.6 and 5.2. The combination of systems and equipment assures that the reactor coolant pressure boundary design conditions are not exceeded by normal operation including anticipated operational occurrences.

  The reactor coolant pressure boundary integrity will not be adversely affected by the application of qualified Alloy 690 sleeves. SG tube surveillance requirements continue to ensure that defective tubes will be repaired or removed from service upon detection. The combination of systems and equipment assures that the reactor coolant pressure boundary design conditions are not exceeded by normal operation including anticipated operational occurrences.

- GDC 19, *Control Room*, which requires a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including Loss of Coolant Accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 Rem whole body, or its equivalent to any part of the body, for the duration of the accident.

  Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

  Conformance: The control room habitability systems include radiation protection, air purification, climatically controlled ventilation and air conditioning systems, lighting and power systems. Collectively, these habitability systems ensure that the control room operators can remain in the control room and take action to operate the plant safely under normal conditions and to maintain it in a safe condition under all accident conditions including a safe shutdown earthquake (SSE) or design tornado.

  The control room heating, ventilation, and air conditioning (HVAC) system functions to prevent air in-leakage during normal and post-accident conditions by filtration of airborne radioactive iodines in the control room atmosphere and also has the ability to purge the room of smoke in the event of a fire. Loss of offsite power will not impair the system's ability to perform its function.

  The original design basis for control room shielding was to limit whole-body dose to plant personnel to less than 25 rem after 30 days following a Design Basis Accident (DBA). As a result of NUREG 0737 Item III.D.3.4 and II.B.2.2, Licensing Bases were changed, Control Room doses were reanalyzed, and HVAC was modified. The current Licensing Basis for Control Room habitability is to assure that occupancy can be maintained under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body, for the duration of an accident.

  The sleeve was qualified under normal operating and faulted conditions to maintain the pressure boundary. A single postulated tube rupture event is already evaluated in the design basis for the plant safe shutdown under normal operating or faulted conditions. Postulated leakage from sleeving is considerably less than the bounding faulted condition leakage necessary to ensure that control room doses remain within 10 CFR 50 Appendix A, GDC 19 limits.

- GDC 30, *Quality of Reactor Coolant Pressure Boundary*, which requires that components which are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

  Conformance: The quality of the reactor coolant pressure boundary meets Criterion 30 as discussed in UFSAR Section 5.1.5.1.

  As discussed above in Section 3.4.2, the sleeves are designed and constructed in accordance with ASME B&PV Code Section III, Subsection NB. The sleeve design was qualified by testing for leakage in accordance with ASME B&PV Code Section XI, IWA-4725.

  The quality of the reactor coolant pressure boundary is preserved by sleeving. Sleeve material will be provided to ASME Code requirements. Installation of the sleeves will be via use of quality procedure steps. Therefore, the reactor coolant pressure boundary will not be adversely affected by the application of qualified Alloy 690 sleeves. SG tube surveillance requirements continue to ensure that defective tubes will be repaired or removed from service upon detection.

- GDC 31, *Fracture Prevention of Reactor Coolant Pressure Boundary,* which requires that components which are part of the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

  Conformance: As discussed in UFSAR Section 5.1.5.2, the PCS (also referred to as the RCS) pressure boundary design meets this criterion by the following:

  1. Brittle failure should not occur if the peak stresses do not exceed the yield stresses in the brittle fracture range. The establishment of temperature-pressure limitations for operation below Nil Ductility Transition (NDT) temperature + 60 °F is based on not exceeding yield for the peak stresses. PNP meets the requirements of 10 CFR 50, Appendix G, *Fracture Toughness Requirements* (May 1983), as amended November 6, 1986, for protection against nonductile failure.

  2. Stress limitations are used to establish pressure-temperature operating curves for the plant. The stress limitations are reflected in plant heatup and cooldown rates and in inservice leak rate testing. The pressure-temperature operating curves consider heatup and cooldown in both critical and noncritical reactor conditions. Protection

against overpressurization of the PCS as a function of coolant temperature has been provided.

3. Quality control procedures include permanent identification of materials and nondestructive testing for flaw identification.

4. Operating restrictions are to prevent failure resulting from increase in brittle fracture transition temperature due to neutron irradiation, including a material irradiation surveillance program. PNP meets the requirements of 10 CFR 50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements,* with regard to reactor surveillance programs. Additionally, PNP complies with the requirements of 10 CFR 50.61a, *Alternate fracture toughness requirements for protection against pressurized thermal shock events,* for protection against pressurized thermal shock events.

5. The primary coolant pressure boundary components were designed, fabricated, erected and tested to the highest quality standards available at that time. The codes utilized were ASME, Section III, including all addenda through Winter 1965 and the code for pressure piping, American Standards Association (ASA) B31.1, 1955.

As discussed in Sections 3.5 and 3.9, the Alloy 690 tube support plate sleeve was designed to meet the ASME Code and restores the Alloy 600 tube to its original design function. Therefore, the PCS pressure boundary integrity will not be adversely affected by the application of qualified Alloy 690 sleeves.

- GDC 32, *Inspection of Reactor Coolant Pressure Boundary*, which requires that components which are part of the reactor coolant pressure boundary be designed to permit: (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

  Conformance: The inspection of the reactor coolant pressure boundary meets Criterion 32 as discussed in UFSAR Section 5.1.5.3. The primary coolant pressure boundary design meets this criterion since space has been provided to permit nondestructive testing of critical areas during unit shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 has been established. In addition, the Plant performs inspections according to the Inservice Inspection Program pursuant to 10 CFR 50.55a, *Codes and standards*.

  The sleeve installation in the parent tube defect region as well as the remaining un-degraded tube are required to be inspected by site approved versions of ASME Code Section XI Subsection IWA 4725 and ASME Code Section V. Defects that may challenge the integrity in maintaining the reactor coolant pressure boundary not satisfying plant TS are plugged at both ends of the tube thus re-establishing the pressure boundary integrity.

Therefore, the reactor coolant pressure boundary integrity will not be adversely affected by the application of qualified Alloy 690 sleeves. SG tube surveillance requirements continue to ensure that defective tubes will be repaired or removed from service upon detection.

In conclusion, conformance with the above GDCs remains valid.

4.6 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

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, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

The PNP operational and support activities quality assurance program is discussed in UFSAR Section 15.1.

4.7 10 CFR 50.55a, Codes and Standards

Part 50.55a of 10 CFR requires, in part, that reactor coolant pressure boundary components meet the requirements for Class 1 components in Section III of the ASME Code. Part 50.55a further requires, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, ASME Code Class 1 components 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.

As discussed in Section 3.9, the design and qualification of the sleeve is governed by applicable industry codes and standards as summarized in Table 3.9-1. As discussed in Section 3.5, the TSP sleeve meets the design requirement of maintaining structural adequacy during normal operating and accident conditions.

4.8 Regulatory Guide 1.121

RG Guide 1.121 provides guidelines for determining the degradation limits for PWR SG tubes. Since the sleeve replaces the original tube, these guidelines are used to determine the plugging limits for the sleeve.

Compliance with the guidelines of RG 1.121 is discussed in Section 3.5.1.

5.0 PRECEDENT

The most recent sleeving amendment was for Watts Bar Nuclear Plant, Unit 2.

By letter dated September 30, 2019 (ML19274C003), as supplemented by letters dated November 21, 2019, and April 6 and June 12, 2020 (ML19326C091, ML20098D779, and ML20164A237, respectively); the Tennessee Valley Authority, submitted an LAR for changes to the Watts Bar Nuclear Plant, Unit 2 (WBN Unit 2) TSs. The requested changes revised the TS to allow the use of Westinghouse non-nickel banded Alloy 800 sleeves to repair defective SG tubes as an alternative to plugging the tubes. The NRC issued the Safety Evaluation on August 10, 2020 (ML20156A018).

While the above does not reflect the configuration to be used at PNP, the information submitted to obtain the Watts Bar Unit 2 approval was used as guidance in the development of this request.

6.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In accordance with 10 CFR 50.92, *Issuance of amendment,* Holtec has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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.

Holtec proposes to revise the Appendix A Permanently Defueled Technical Specifications (PDTS) of the Palisades Nuclear Plant (PNP) Renewed Facility Operating License (RFOL). The proposed changes are based on the reinstatement of the plant operating licensing basis (POLB), as described in Revision 35 of the PNP Updated Final Safety Analysis Report (UFSAR), and the resumption of power operations at PNP.

The proposed license amendment revises the PNP Technical Specifications (TSs) proposed in Reference 1 to permit 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 discussion below addresses each 10 CFR 50.92(c) no significant hazards consideration criterion and demonstrates that the proposed amendment does not constitute a significant hazard.

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.

Based on the above, Holtec concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

Conclusion

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

7.0 ENVIRONMENTAL EVALUATION

Consistent with NRC guidance and the No Significant Hazards Consideration in Section 6 above the proposed amendment permits the use of Framatome Alloy 690 TSP sleeves to repair defective SG tubes as an alternative to removing the tubes from service by plugging.

i) The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated,
ii) The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated, and
iii) The proposed amendment does not involve a significant reduction in a margin of safety.

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

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

8.0 REFERENCES

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

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

3. NRC letter to HDI, "Palisades Nuclear Plant and Big Rock Point Plant – Issuance of Amendment Nos. 129 and 273 re: Order Approving Transfer of Licenses and Conforming Administrative License Amendments (EPIDS L-2022-LLM-0002 AND L-2020-LLM-0003)," dated June 28, 2022 (ADAMS Accession No. ML22173A173)

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

5. NRC letter to HDI, "Palisades Nuclear Plant – Summary of Conference Call Regarding Steam Generator Tube Inspections (EPID L-2024-NFO-0008)," dated October 1, 2024 (ADAMS Accession No. ML24267A296)

6. HDI letter to NRC, “Application for Order Consenting to Transfer of Control of License and Approving Conforming License Amendments,” dated December 6, 2023 (ADAMS Accession No. ML23340A161)

7. HDI letter to NRC, “Final Safety Analysis Report Update - Revision 35,” dated April 14, 2021 (ADAMS Accession No. ML21125A285)

Enclosure 2

PNP 2025-003

Technical Specification Page Markups

9 pages follow

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 $\geq$ 90% RTP. --- Verify PCS total flow rate within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program AND After each plugging or repairing 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, ~~or~~ 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 Programs and Manuals

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."

between the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, and

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

12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower,

1. Tubes found by inservice inspection to contain service induced flaws ~~within~~ 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 ~~this elevation~~ may remain in service.

between the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and

2. Tubes found by inservice inspection to contain service induced flaws ~~within~~ 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 13.67 inches below ~~this elevation~~ may remain in service.

the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower,

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 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

5.5.8 Steam Generator (SG) Program (Continued)

d. Provisions for SG tube inspections. (continued)

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 ~~applicable tube~~ 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.

(Insert from next page.)

(Insert at the bottom of Page 5.0-12)

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.

5.6 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, ~~and~~

h. The effective plugging percentage for all tubes plugged or repaired ~~plugging~~ 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 3

PNP 2025-003

Retyped Technical Specification Pages

Note: Enclosed Technical Specification page 5.0-14 and 5.0-15 do not contain changes. They are included for completeness due to previously added text resulting in text being rolled to these pages.

11 pages follow

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 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. The following SG alternate repair criteria shall 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 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 Programs and Manuals

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 $\leq 30°C$ 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 4

PNP 2025-003

Technical Specifications Bases Mark-Ups
(For information only)

5 pages follow

BASES

ACTIONS

A.1

Pressurizer pressure and cold leg temperature are controllable and measurable parameters. PCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. With any of these parameters not within the LCO limits, action must be taken to restore the parameter.

The 2-hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits.
The Completion Time is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 and SR 3.4.1.2

The Surveillance for monitoring pressurizer pressure and PCS cold leg temperature is performed using installed instrumentation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.1.3

Measurement of PCS total flow rate verifies that the actual PCS flow rate is within the bounds of the analyses. This verification may be performed by a calorimetric heat balance or other method.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. PCS flow rate must also be verified after plugging **or repairing the number** of ~~each 10 or more~~ steam generator tubes **which results in the same primary flow reduction as plugging 10 tubes** since plugging 10 or more tubes could result in an increase in PCS flow resistance. Plugging **or repairing** less than **the equivalent of** 10 steam generator tubes will not have a significant impact on PCS flow resistance and, as such, does not require a verification of PCS flow rate.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate that bounds the operational LEAKAGE rate limits in LCO 3.4.13, "PCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via the Main Steam Safety Valves and Atmospheric Dump Valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.3 gpm or is assumed to increase to 0.3 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "PCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the applicable limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), 10 CFR 50.67 (Ref. 7) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the **plugging or** repair criteria be plugged **or repaired** in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program **plugging or** repair criteria is **repaired or** removed from service by plugging. If a tube was determined to satisfy the **plugging or** repair criteria but was not plugged **or repaired**, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall **and any repairs made to it**, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "PCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

PCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube **plugging or** repair criteria but were not plugged **or repaired** in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG **plugging or** repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged **or repaired** has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is

ACTIONS (continued)

A.1 and A.2 (continued)

discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged **or repaired** prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.1 (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube **plugging or** repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program **plugging or** repair criteria is **repaired or** removed from service by plugging. The tube **plugging or** repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube **plugging or** repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the **plugging or** repair criteria are plugged **or repaired** prior to subjecting the SG tubes to significant primary to secondary pressure differential.

Enclosure 5

PNP 2025-003

Framatome Document Number 51-9388710-001, "Steam Generator Mechanical TSP Sleeve Qualification Assessment for ¾" Tubes at Palisades Nuclear Power Plant" (Non-Proprietary) (Non-Proprietary version of Framatome Document Number 51-9385467-002)

59 pages follow

Framatome Inc.

Engineering Information Record

Document No.: 51 - 9388710 - 001

Steam Generator Mechanical TSP Sleeve Qualification Report for ¾" Tubes at Palisades Nuclear Power Plant

~~**EXPORT CONTROLLED INFORMATION**~~

~~Contains technology subject to U.S. export controls governed by the Export Administration Regulations (15 CFR Part 730 et seq.) and/or the Department of Energy Regulations (10 CFR Part 810). Diversion contrary to U.S. law is prohibited.~~

~~**Export Classification**~~	~~**US EC:**~~ ☐ ~~N~~ ☐ ~~Part 810~~ ☒ ~~EAR~~	~~**ECCN:** 2E290~~

Steam Generator Mechanical TSP Sleeve Qualification Report for ¾” Tubes at Palisades Nuclear Power Plant

Safety Related? ☒ YES ☐ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title	Signature and Date	Role	Scope/Comments
Richard Coe Consulting Engineer		LP	All
Steve Cook Advisory Engineer		LR	All
Wayne Belden Manager, SG Performance Engineering		A	All

Role Definitions:

P/R/A designates Preparer (P), Reviewer (R), Approver (A);
LP/LR designates Lead Preparer (LP), Lead Reviewer (LR);
M designates Mentor (M);
PM designates Project Manager (PM)

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Original Issue. Proprietary version is 51-9385467-001.
001	Section 7 and Section 11	Revised Section 7.7 and Deleted Section 11 "No Significant Hazards and Environmental Considerations" Proprietary version is 51-9385467-002.

Table of Contents

Table of Contents
(continued)

List of Tables

Page

List of Figures

Page

1.0 EXECUTIVE SUMMARY

Framatome has designed and qualified a mechanical hydraulically expanded Alloy 690 sleeve for repair of Palisades steam generator degraded tubes at lattice tube support plates (TSPs). The sleeve design has two sets of hydraulic expansions above and below the TSP. There is also a third small hydraulic expansion in the middle of the sleeve that imparts a [ ] to the sleeve outer joints thus improving the leak rate performance to the outer hydraulic expansion joints. The sleeve is fabricated from thermally treated Alloy 690 material providing enhanced corrosion resistance compared to the steam generator tubing.

The sleeve qualification and testing comply with the in-service Inspection and Repair requirements of ASME Code Section XI, IWA-4721, and more specifically IWA-4725 pertaining to the Expansion Requirements for sleeves, 2007 Edition with Addenda through 2008.

The analytical evaluations were originally completed to qualify the Framatome sleeves for use in Westinghouse [ ] steam generators. Where necessary, the qualification has been extended by analysis to provide a qualified TSP sleeve for use at Palisades.

The sleeve qualification included axial load, pressure, and temperature cycling. The sleeves were axially cycled for 10 years' worth of plant transients and thus have been qualified beyond Palisades' currently licensed plant life. With further [ ] life can be addressed to cover a Palisades life extension.

Fatigue loadings for use in mechanical qualification testing of the sleeve and tube joint were calculated per the ASME B&PV Code. The highest loadings from these conditions were conservatively used in the mechanical load testing.

A stress and fatigue evaluation were performed consistent with Regulatory Guide 1.121 and NEI 97-06 [1] to establish the plugging criteria for the sleeve in the expansion regions. The limiting case was the burst pressure criteria during normal operation for the joint region of the sleeve. This resulted in a calculated acceptable flaw depth for the sleeve of [ ] through wall. NEI 97-06 [1] allows for non-destructive examination (NDE) detection and sizing uncertainties and possible defect growth between inspections. The results of this evaluation validate that flaws [ ] TW satisfy all performance criteria including NDE uncertainty and flaw growth. In addition, the Palisades technical specification tube plugging limit of $\leq$40%TW is also applicable to the TSP sleeve. The Palisades Steam Generator Program currently requires plugging of crack-like indications on detection in the parent tubing, except in areas within the tubesheet covered by an alternate repair criterion. This same plug-on-detection criteria applies to the parent tubing portion of the pressure boundary in the sleeve joint regions. SG tube surveillance requirements continue to ensure that defective tubes will be repaired or removed from service upon detection.

From a thermal hydraulic standpoint, the sleeve has a smaller impact on plant operations compared to plugging that same tube. Installing [ ] Sleeving is a good option to maintain the effective plugged tube quantity below the plant's design plugging limit.

Installation of sleeves in both the hot and cold leg tube support locations of the same tube will impede the ability to efficiently inspect the un-sleeved section of the tube. Since most SCC indications are on the hot leg, eddy current inspection from the cold leg through the U-bend is facilitated.

2.0 INTRODUCTION

2.1 PURPOSE

The purpose of this report is to document the qualification of the Framatome hydraulically expanded Alloy 690 Tube Support Plate (TSP) sleeve in Recirculating Steam Generators (RSGs) with nominal 0.750 in. OD x 0.042 in. Alloy 600 tubes for the Palisades Nuclear Power Plant. This report includes sufficient information to support a technical specification change allowing installation of these sleeves and demonstrates that reactor operation with sleeves installed in the Palisades RSGs will not increase the probability or consequence of a postulated accident condition that has been previously evaluated. The sleeve installation will not create the possibility of a new or different kind of accident and will not reduce the existing margin of safety.

The qualification program described in this report includes information on the design analyses, leak and cyclic testing, corrosion resistance evaluation, nondestructive examination, the installation process, and ALARA aspects of the sleeve design. This report encompasses [ ] with four hydraulically expanded joints on both ends of the sleeve, above and below the defective region of tube. The sleeves are acceptable to be installed within any tube support plate except elevations that are geometrically limiting at the Palisades Nuclear Power Plant.

2.2 BACKGROUND AND EXPERIENCE

Framatome US has installed 23,933 sleeves in steam generators and other heat exchangers. (See Figure 2-1) Framatome first installed sleeves in a nuclear steam generator in 1978. Between 1978 and 1997, Framatome US has installed 10,682 sleeves in both recirculating and once through steam generator designs in the United States and Europe. The steam generator sleeve encompassed many different attachment joint designs including explosive welded, brazed, roll expanded, and hydraulically expanded. None of the 10,682 steam generator sleeves are currently in service because those steam generators were replaced, or the plant was retired.

Due to steam generator replacement, the market for sleeving shifted to other heat exchangers and Framatome continued to install 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 proposed for installation in the Palisades steam generator was designed, developed, and qualification tested in the 1994 timeframe just as the steam generator market was shrinking. This design, consisting of eight hydraulically expanded regions (H8), [ ] has been installed 6,007 times in non-steam generator heat exchangers. Of the 6,007 H8 sleeves installed, the H8 design has been installed 964 times in ASME Section III heat exchangers including CCW and containment spray heat exchangers. Framatome is not aware of any issues with the sleeves installed in the non-steam generator applications. The sleeves installed in the ASME Section III heat exchangers fall under 10CFR Part 21 Reporting of Defects and Noncompliance and Framatome has received no notifications regarding 10CFR Part 21 pertaining to these sleeves.

Figure 2-1: Framatome Sleeve Installation Experience

3.0 ASME CODE DISCUSSION

The design and qualification of the sleeve is governed by applicable industry codes and standards as summarized in Table 3-1: . The ASME B&PV code is the basic governing document for numerous aspects of the design, including determining test loads, performing structural analyses, procuring material, establishing the sleeve procedure qualification, and preparing the sleeving procedure specification. In all cases, the mechanical TSP sleeve meets the requirements of the ASME B&PV code.

The sleeves are designed and constructed in accordance with ASME B&PV Code Section III, Subsection NB [2]. The sleeve design and procedure were qualified by testing for leakage in accordance with ASME B&PV Code Section XI, IWA-4725 [2].

The sleeve qualification considered temperatures, pressures, and cyclic testing enveloping plants having steam generators with ¾ in. tubing. The design specification [3], prepared specifically for Palisades, specifies loading, geometry, and materials specific to the Palisades RSGs. Reconciliation is performed herein to demonstrate that the qualification testing and analysis for the sleeves uses the same or bounding loading, geometry and materials specified in the design specification [3].

The repair activities are completed in accordance with the NRC approved version of the ASME B&PV 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 NRC approved version of ASME Section XI, 1989 Edition with no Addenda and supplemented with ASME Section XI, 2007 Edition with Addenda through 2008 [2]. 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 Section in the 2007 Edition with 2008 Addenda of the Code is IWA-4725, Expansion. A comparison between the Code Editions was completed and it is concluded that while there are administrative changes that do not require reconciliation between the various Editions of the Code, there are no changes to technical requirements between the Code Editions. Therefore, the technical requirements of the ASME B&PV Code, Section XI, 2007 Edition with Addenda through 2008 are met.

The materials, design, fabrication, and examination of the Palisades steam generator tubes meet all the requirements of ASME B&PV Code, Section III, 1977 Edition [2]. The materials, design, fabrication, and examination of the sleeves meet all the requirements of ASME Section XI, 1989 Edition and the 2007 Edition with Addenda through 2008. Use of later code is permitted by ASME Section XI, IWA-4221, provided reconciliation is performed in accordance with IWA-4222 through IWA-4226 as applicable. Those paragraphs require the sleeves be evaluated for changes to the weight, configuration, and pressure-temperature rating of the steam generators.

This report demonstrates the acceptability of the sleeve installation regarding the weight, configuration, and pressure-temperature ratings of the steam generators and meets the requirements of IWA-4222 through IWA-4226.

Table 3-1: Summary of RSG Sleeve Applicable Codes and Standards

Application	Criteria
Structural Design of the Sleeve	ASME B&PV Code Section II & III
Sleeve Plugging Limit	NRC Reg. Guide 1.121 NEI 97-06
Material Procurement	ASME B&PV Code Section II & III
Mechanical Sleeve Qualification	ASME B&PV Code Section XI
Sleeve NDE	ASME B&PV Code Section V & XI

4.0 ACCEPTANCE CRITERIA

The objective of installing sleeves in steam generator tubes is twofold. The sleeves must maintain structural integrity of the steam generator tube during normal operating and postulated accident conditions and the sleeve must limit the primary to secondary leakage in the event of a through wall defect in the section of the steam generator tube spanned by the sleeve. Numerous tests and analyses were performed in the 1994 timeframe to demonstrate the capability of the Framatome mechanical TSP sleeve for Westinghouse [ ] steam generators and a 10-year life. An applicability assessment [4] has been completed to demonstrate that the numerous tests and analyses also bound the Palisades Nuclear Power Plant steam generators. Palisades is bounded by the following conservative design basis values from WEC [ ] plants. While the absolute values may differ, the evaluations are a function of differential pressures and temperatures.

Table 4-1: Bounding Design Basis Values

Parameter	[ ]	Palisades [3]
Design Primary Side Temperature		650°F
Design Secondary Side Temperature		550°F
Design Primary Pressure		2500 psi
Design Secondary Pressure		1000 psi
Maximum Differential Pressure (Normal conditions)		1500 psi

To qualify the sleeves, a set of sleeve criteria has been established to demonstrate the acceptability of the Alloy 690 sleeve for Palisades. The criteria and justification for this review are provided in Table 4-2. The results column provides a reference to appropriate section (provided in Table 4-2) that provides the justification that the criteria has been met for Palisades.

Table 4-2: Sleeve Criteria

Criterion	Approach	Results	Section
Exposure of the repair assembly to various primary and secondary chemistries without a loss of function	Literature research and operating experience for Alloy 690 material.	Alloy 690 is extremely resistant to the chemical environments present in PWR primary and secondary water chemistries.	6
Repair sleeve-tube assembly structural integrity must be maintained for normal and accident conditions	Repair sleeve-tube assembly meet applicable ASME Code requirements, including fatigue.	Framatome sleeve meets the requirements of ASME Code including Fatigue. Results are documented in Table 7-1	7
Sleeve/tube joint load capability 3 times normal Δp and 1.4 times steam line break Δp even for a severed tube. Does not collapse sleeve during LOCA	RG1.121 and NEI 97-06 plugging criteria analysis	Sleeve plugging criteria established to meet 3 times normal Δp and 1.4 times steam line break Δp even for a severed tube.	7
Sleeve/Tube Joint deflection capability is sufficient for thermal expansions effects.	No degradation of leak limiting or structural load capability for worst case thermal expansion cycles.	No sleeve motion or degradation was noted from cycle testing	8
Pressurization of annulus between the sleeve and tube does not fail during operation or design basis accidents	Prevention of sleeve failure based on pressure testing	No sleeve motion or degradation as noted from cycle testing	8
Repair sleeve installation to satisfactorily limit leakage in any direction and under normal and accident conditions.	Allowable leakage established by Palisades technical specifications.	Leak rate estimates per sleeve.	8
Repair sleeve effect on system flow rate and heat transfer capability of the steam generator is acceptable.	Allowable reduction in coolant flow rate is limited per Palisades technical specifications. Steam pressure reduction due to reduced heat transfer is limited for commercial considerations.	[ ]	7
Non-destructive examination of the tube and sleeve pressure boundary with levels of detectability sufficient to show structural adequacy.	Periodic examination of tubes and sleeves are required.	Eddy current inspection capability is demonstrated as capable of detecting degradation within a sleeved tube.	10

5.0 DESIGN DESCRIPTION OF SLEEVES

Sleeving is a method used to repair defective steam generator tubes and thus keep the tubes in service. A lattice tube support plate (TSP) sleeve is designed to repair tube degradation occurring at TSP intersections of SG tubes. A mechanical sleeve is a tube segment that is inserted into an existing SG tube and expanded to create an interference fit between the sleeve and the tube. The proposed sleeve uses hydraulic expansion to create the interference fit

The ¾ in. mechanical TSP sleeve is fabricated from tubing with a [ ] minimum wall thickness in the unexpanded region. The tubing material for the sleeve is thermally treated Alloy 690 (UNS N06690) ordered in accordance with Reference [2]. Section 6.1 discusses the basis for selecting Alloy 690 as the sleeve material.

[ ] This evaluation considered structural issues within the steam generator as well as the defect locations with respect to the edges of the TSPs.

Figure 5-1 illustrates some of the locations that cannot be sleeved due to interference with the bend in the tube or due to placement of the upper joint at a u-bend diagonal support location. Figure 5-2 shows a typical installation.

The free span sleeve/tube joints are produced by a [ ] controlled hydraulic expansion process. A total of eight joints are produced simultaneously, with four above and four below the TSP.

The sleeve is fabricated with OD sealing ribs on all eight of the expansion areas. The profile of these ribs, when combined with the sleeve expansions, gives the sleeve improved leak limiting characteristics compared to the expansion of a smooth sleeve. Refer to Figure 5-3 for a longitudinal section of the sleeve.

The structural and pressure boundaries are defined as the load carrying portion of the installed sleeve. This is the non-sleeved tube, the portion of the tube covering the eight hydraulic expansion joints, and the sleeve from the outboard portion of the outermost expansion to the opposite end of the sleeve. This is shown in Figure 5-4.

The variations in tube OD expansion due to hydraulic expansion [ ]

Following the hydraulic expansion of the eight areas, a [ ] controlled hydraulic expansion is made [ ] allowing the sealing ribs to perform their function more effectively. This partial expansion is performed such that the sleeve does [ ] The expansion length is set to achieve the [ ] regardless of sleeve length.

Figure 5-1: TSP Locations that Cannot Be Sleeved

Figure 5-2: Typical Sleeve Installation

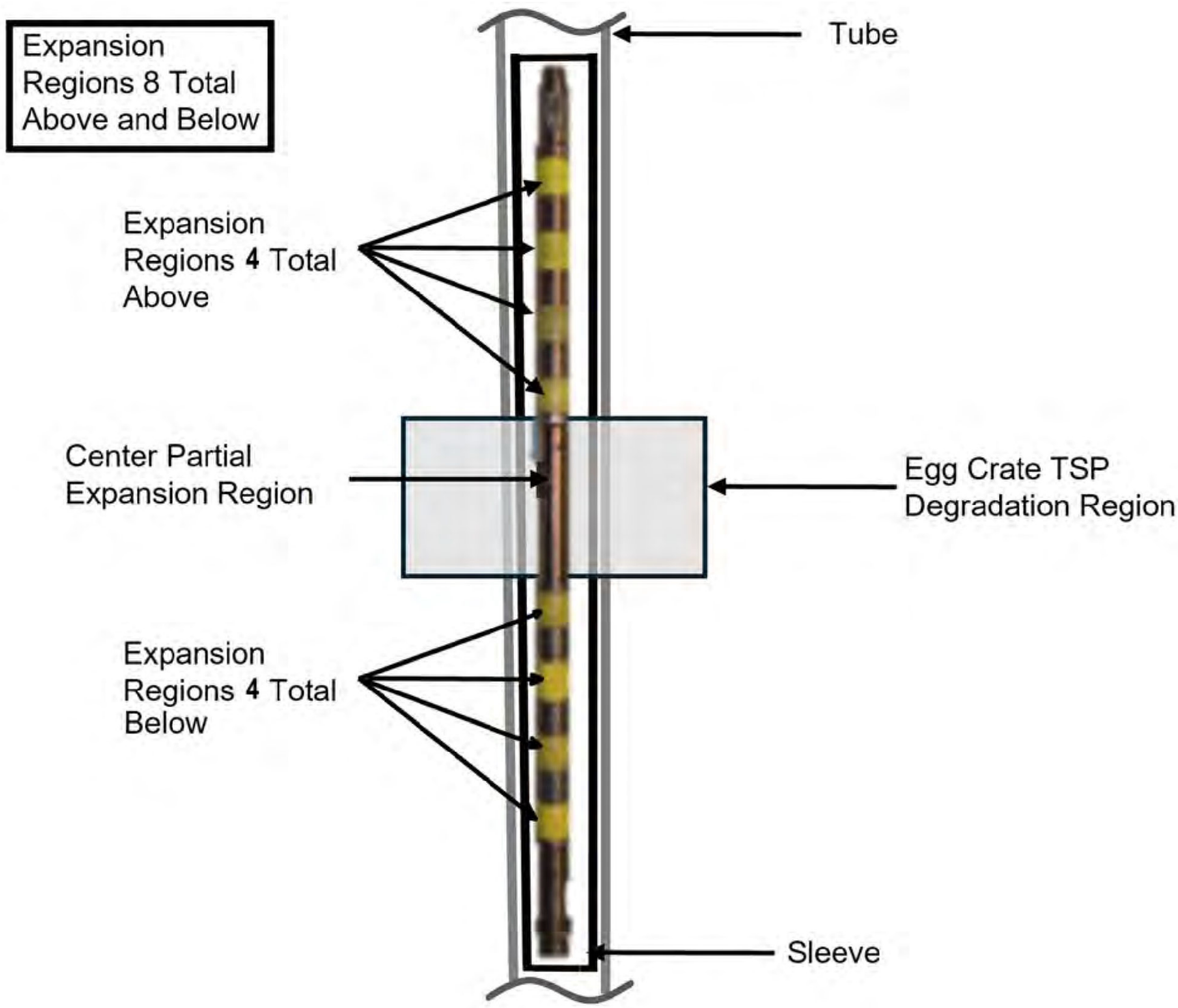


Figure 5-3: Mechanical TSP Sleeve

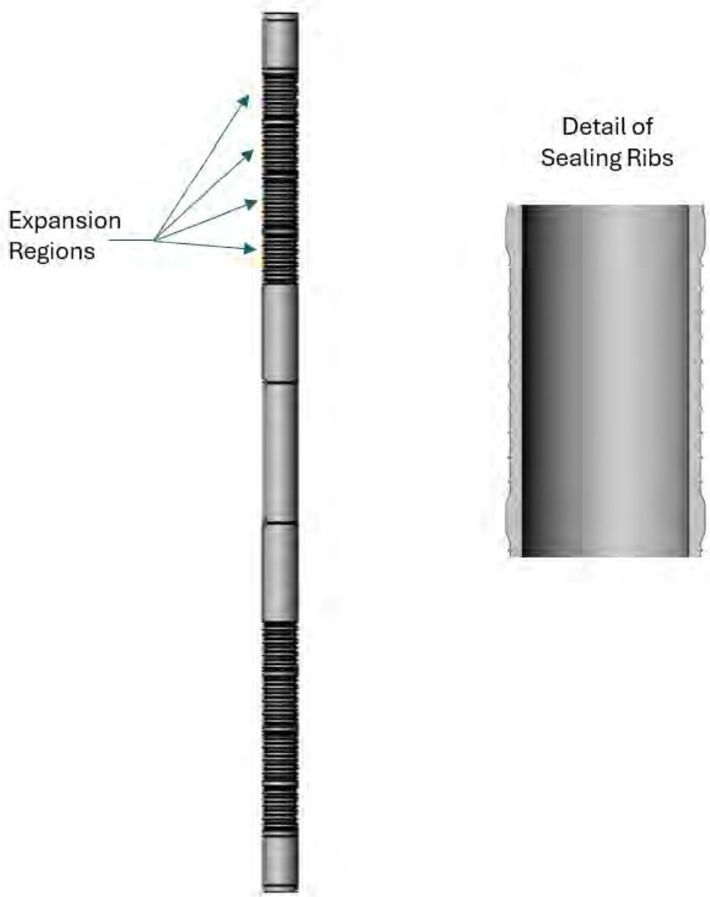


Figure 5-4: Sleeve Structural and Pressure Boundary Description

[

]

6.0 CORROSION EVALUATION

After the introduction of all-volatile chemistry control, the main corrosive attacks on Alloy 600 tubing can be divided into two forms of stress corrosion cracking (SCC):

- Pure Water (or Primary Side) SCC (or PWSCC)
- Secondary Side SCC (or ODSCC)

This section will summarize:

- The details that drive these two forms of SCC and the locations where these corrosion mechanisms are seen in the steam generator tubing.
- The susceptibility of the sleeve material (Alloy 690) to these degradation mechanisms.
- The potential impact of sleeve installation on further tube degradation.

6.1 Sleeve Material Selection

The mechanical TSP sleeve material is thermally treated Alloy 690. The selection criteria for the sleeve material were mainly led by its excellent corrosion resistance in both primary side and secondary PWR environments.

With the significant problems occurring in Alloy 600 tubing, a world-wide effort was put into developing a new alloy for steam generator tubing. The two potential candidates, Alloy 690 and Alloy 800, were subjected to various test environments to determine their overall corrosion resistance. In the US, the material chosen for tubing in replacement steam generators was overwhelmingly Alloy 690.

Some replacement SGs in Europe as well as some CANDU reactor designs used Alloy 800 tubing in their SGs. Years after SG replacements were common, a 2009 EPRI Report [6] concluded that Alloy 690 offered greater corrosion resistance in SG environments (from testing and in-service) than Alloy 800.

With decades of experience with replacement SGs, the conclusions from testing that Alloy 690 was virtually impervious to PWSCC and greatly improved to ODSCC have been proven. In some SGs with Alloy 800 tubing, ODSCC has been found in tubesheet crevices, within lattice support plate locations, and at dents [7]. Thus Alloy 690 has proven to be the best alloy currently available for both primary and secondary side corrosion concerns.

6.2 Alloy 690 Operating Experience

The principal concern to the sleeve joint on the primary side is the potential for primary water stress corrosion cracking (PWSCC) because of the stresses imparted to the tube due to the sleeve installation. PWSCC is not a principal concern for Alloy 690 due to the superior resistance to PWSCC. Estimates put the factor of improvement of Alloy 690 over Alloy 600 at 40 to 100 times for crack initiation [8]. In addition, a significant number of plants both domestically and internationally have replaced steam generator with Alloy 690 tubes. Table 6-1 contains a list of plants that have replacement steam generators as documented in Reference [8] through 2008. A significant portion of the U.S. PWR fleet have replaced steam generators with Alloy 690 tubes and continue to demonstrate excellent operating experience and resistance to PWSCC. Alloy 690 and its weld metals have been used in most PWR replacement component items, such as steam generator tubing, repair sleeves, plugs, pressurizer heater sleeves, and vessel closure head nozzles. Alloy 690 has been in service since the late 1980's and no cracking of the Alloy 690 materials have been observed in the U.S. or international plants [9].

Table 6-1: Plants with Alloy 690 Steam Generator Tubing in Replacement S/Gs

Installation Year	PWR
1989	D.C. Cook-2, Indian Point-3, Ringhals-2
1990	Dampierre-1
1991	Ohi-3
1992	Penly-2
1993	Milestone-2, North Anna-1, Beznau-1, Ohi-4
1994	Mihama-2, Takahama-2, Genkai-1, V.C. Summer, Gravelines-1, Daya Bay-1, Daya-Bay-2, Golfech-2, Genkai-3, Ikata-3
1995	Ohi-1, Tihange-1, North Anna-2, Ringhals-3, Dampierre-3, St. Laurent-B1, Sizewell-B
1996	Mihama-1, Takahama-1, Doel-4, Catawba-1, Ginna, Gravelines-2
1997	McGuire-1&2, Point Beach-2, Mihama-3, Ohi-2, Tricastin-2, Genkai-4
1998	Byron-1, St. Lucie-1, Tihnage-3, Braidwood-1, Kori-1, Ikata-1, Tricastin-1
1999	Beznau-2
2000	Farley-1, D.C. Cook-1, Krsko, STP-1, ANO-2, Chooz-B1, Civaux-1&2, Gravelines-4
2001	Genkai-2, Ikata-2, Kewaunee, Shearon Harris, Farley-2, Tihange-2
2002	Tricastin-3, Fessenheim-1, Chooz-B2, Calvert Cliffs-1, STP-2
2003	Calvert Cliffs-2, Palo Verde-2, Sequoyah-1, Oconee-1, St-Laurent Des Eaux B
2004	Tricastin-4, Oconee-2, Prairie Island-1, Oconee-3, Dampierre-2
2005	Callaway, ANO-1
2006	Beaver Valley 1, Fort Calhoun, Watts Bar-1
2007	Comanche Peak 1, St. Lucie-2
2008	Diablo Canyon 2

6.3 PWSCC of Alloy 600

PWSCC requires three synergistic elements (see Figure 6-1) to occur:

1. Tensile stress (both operational and residual),
2. An aggressive (corrosive) environment,
3. A susceptible material.

Figure 6-1: Conditions Required for SCC

Industry experience has demonstrated that the primary side environment is sufficiently aggressive and that Alloy 600 in all heat-treat conditions (low-temp mill anneal, high-temp mill anneal, and thermally treated) is susceptible to PWSCC when subjected to tensile stresses. Industry-wide, PWSCC has been seen at:

- Tubesheet expansion transitions
- Anomalies within the tubesheet expansion
- U-Bend region (especially in tubing with the smallest bend radius)
- Dent/Ding locations in the tubing.

6.4 ODSCC of Alloy 600

Alloy 600 PWR steam generator tubing has been affected by SCC issues from the secondary side due to impurity concentration by boiling in superheated crevices, usually at tube supports and under corrosion product sludge piles that can accumulate on tubesheet. This cracking phenomenon is accepted to be a completely different mechanism to that occurring on the PWR primary side. At Palisades, ODSCC has been detected at the top of the hot leg tubesheet, lattice support plates (predominately on hot leg), as well as a smaller number of indications at diagonal bars and vertical straps.

6.5 Impact of Sleeve Installation on Alloy 600 Tubing

The installation process plastically deforms the sleeve and the tube by performing eight simultaneous hydraulic expansions, securing the sleeve above and below defects at the lattice support plate locations. Thus, an evaluation is required to evaluate the effect of the residual stresses on the service life of installed sleeve. Because of the excellent corrosion resistance of the Alloy 690 used to make the sleeves, Framatome's investigation focused on the effect the residual stresses from sleeve installation would have on the Alloy 600 tubing.

Framatome’s approach to evaluating the effect of the residual stresses from sleeving in Alloy 600 tubing was to:

1) Collect industry data on the various expansion techniques used to expand tubing into the tubesheets during SG manufacturing.
2) Perform testing and analysis of the freespan hydraulic joint made during the sleeving process.
3) Estimate the service life of Framatome’s hydraulic sleeve by making comparisons to applicable processes used for making tube expansions into tubesheets.

6.5.1 Applicable Industry Investigations

1) Testing [10] has shown that the residual tensile stresses for rolled tubesheet expansions range from 50-58 ksi and 36-44 ksi for hydraulic tubesheet expansions.
2) The following conclusions were reached on ID residual stresses from hydraulic tubesheet expansions [11].
    a. The peak residual tensile stresses are axial in nature and occur on the ID surface of the tube.
    b. The peak axial residual stress varies with the room temperature yield strength (RTYS) and are approximately 75-80% of the RTYS.
    c. The peak residual tensile stresses occur near the expanded/unexpanded edge of the transition zone.
    d. The expansion diameter and pressure had little effect on the residual stresses.
3) Explosively expanded tubesheet samples have been corrosion tested to determine their resistance to PWSCC [12]. The corrosion tests showed that the residual stresses range from <30 to as high as 50 ksi. The residual stresses from explosive expansions are similar to those reported from hydraulic tubesheet expansions in [10].

6.5.2 Framatome Testing and Analysis

6.5.2.1 Stress Indexing Test

[ ]

6.5.2.2 X-Ray Diffraction Measurements

The stresses induced by sleeve installation can be measured using X-ray diffraction. The deformation caused by hydraulic expansion results in distortion of the crystal lattice of the alloy 600 parent tubing, which changes the diffraction characteristics of an X-ray beam.

The change in diffraction can be used to calculate the distortion or strain of the crystal lattice. The residual stresses are then calculated from the strain measurements.

[

]

6.5.2.3 Finite Element Analysis

An elastic/plastic finite element model was prepared to model the hydraulic expansion process of sleeve installation. The analysis confirms the peak residual stress [ ] surface of the tube, near the joint transitions where the tube begins to deform. The peak tensile stress [ ]

6.5.2.4 Comparison to Industry Evaluations

Framatome’s investigation into the residual stresses induced by hydraulic expansion of the TSP sleeve agree well with industry evaluation of hydraulic and explosive tubesheet expansions. Specifically:

- [ ]
- The peak residual tensile stresses [ ]
- The peak residual stresses are axial in nature and are [ ]
- Expansion diameter has little effect on the residual stresses induced by hydraulic expansions.

6.5.2.5 Caustic Corrosion Test

[ ] was determined based on a loss of internal pressure.

The following samples were subjected to the caustic corrosion test:

- [ ]
- [ ]

The time to failure for the corrosion samples is contained in the following table

Table 6-2: Caustic Corrosion Test Results

[ ]

The following conclusions were reached from the above test results and examination of the corrosion samples:

- Cracks were [ ] and not in the [ ] are located.
- [ ]
- [ ]

6.5.3 Service Life of Framatome's Hydraulic Sleeve

The residual stresses of the [ ] Thus, the service life of tubing transitions at the secondary face of the tubesheet can be used to estimate when PWSCC would be expected to be found in the tubing at the hydraulic expansions made during sleeve installation. Using the life of the [ ] since the temperature at the tubesheet will be higher than at the lattice support plate locations.

In effect, the operating history of the Palisades SGs serves as a test environment to understand how resistant each individual tube is to PWSCC. For example, if a given tube took [ ]

The operating history of Palisades provides confidence that the hydraulically expanded sleeves will last for the expected remaining life of the SGs.

7.0 DESIGN ANALYSIS

7.1 Sleeve Design Calculations

The sleeve design calculations use allowable stresses based on the strength properties listed in ASME B&PV Section II [2] for Alloy 600 and Alloy 690. The structural adequacy of the RSG sleeves was originally evaluated in Reference [13] for pressure thickness, external pressure, seismic (OBE), and flow induced vibration in accordance with the ASME B&PV Code. An applicability assessment [4] was performed to assess the primary stress qualification and to recalculate the fatigue usage factor for the Palisades specific transients and sleeve condition. [ ] per paragraph NB-3324 of Reference [2]. The specified minimum sleeve wall thickness in the expansion region is [ ] and is therefore conservative. The allowable external pressures for the sleeves and tubes were calculated per paragraph NB-3133.3. The allowable pressure of the sleeve is [ ] than the allowable pressure for the tube. Therefore, the sleeve exceeds the strength of the original tube for external pressure loadings. The analysis of the hydraulic expansion region of the sleeve/tube assembly uses the criteria of subsection NB of the ASME B&PV Code [2] to establish allowable stresses. The allowable values are based on a design temperature of 650°F and are summarized in Table 7-1. The structural integrity of the freespan hydraulic expansion was evaluated for design basis loads using a finite element analysis (FEA) [13]. The expanded sleeve/tube geometry was subjected to a loading condition consisting of internal (primary side) and external (secondary side) pressures which correspond to the design differential pressure conditions. In addition, the stress analysis considered seismic (OBE) loads and flow induced vibration loads. The resulting primary stresses were evaluated per the design requirements of the ASME B&PV Code [2]. The analyzed geometry considered the only contact between the sleeve and the tube to be at the hydraulic expansions and that the tube was severed at the TSP locations. All stresses in the tube and sleeve satisfy the primary stress limits. The secondary-to-primary pressure differential was not limiting. The maximum stress intensities are listed by stress category in Table 7-1. In addition to the stress intensities, fatigue usage for a [ ] is documented for the sleeve and the tube.

7.1.1 Primary Stress Assessment

The primary stress qualification of the sleeve is documented in Reference [13]. Table 7.1 summarizes the results. The primary stress qualification was performed for design and operating conditions; a review of loads and consequent sleeve results concluded that the bounding stress is for a defected sleeve under normal operating conditions.

Table 7-1: Defected and Non-Defected Sleeve and Tube Stresses

[ ]

7.1.2 Assessment of Pressure Load

The effective pressure load (pressure difference between primary and secondary side) used for the sleeve bounding primary stress qualification is [ ] bounds the maximum Palisades [ ] psia, Reference [3].

7.1.3 Assessment of OBE Loads

The OBE load of [ ]

Comparison of the tube support plates spacing which is the main parameter for the tube and sleeve loading indicates that spans of the Palisades lower tube support plates are only [ ] further apart than spans of tube support plates of the original sleeve analysis. The middle and upper Palisades tube support plates are closer than spans of tube support plates of the original sleeve analysis.

The difference in tube support plates spacing is deemed small to result in a significant impact on the sleeve load. The seismic acceleration load of Palisades is significantly lower than the acceleration used for the sleeve qualification. Therefore, Palisades OBE loads is bounded by the sleeve qualification analysis.

7.1.4 Assessment of FIV Loads

[ ] This stress due to FIV only is used for the sleeve primary stress qualification. Per Reference [13], this value represents the peak moment based on standard deviation of 3. The Root Mean Square (RMS) moment value [ ] The Palisades FIV RMS stress [ ] per Appendix D of Reference [16]. The Palisaded FIV load is bounded by the sleeve qualification analysis, Reference [13].

7.1.5 Fatigue Analysis

The prior fatigue analyses were performed for the 3/4" RSG TSP for the Westinghouse [ ] A review of analyzed transients in Reference [13] compared to Palisades design transients [3] concluded that not all Palisades transients are bounded by transients analyzed in Reference [13]. Furthermore, the prior fatigue analyses considered a [ ] condition present in the Westinghouse steam generators, but not present in the Palisades steam generators.

To determine a bounding usage factor of Palisades SG sleeved tube, methodology and unit loads applied in Reference [13] are used. Reference [13] provides stress ranges and fatigue calculation of the bounding location, tube, [ ] outside location of Stress Classification Line (SCL) T1. This fatigue usage factor calculation is fitted for Palisades using the following steps.

1) Apply applicable Palisades transients.
2) Recalculate stress intensity using Palisades primary and secondary pressure and temperatures for each transient.
3) Calculate the axial loads due to pressure.
4) Recalculate transient pairings, stress intensity ranges and number of cycles.
5) Recalculate the penalty factor [ ] based on recalculated stress intensity range
6) Recalculate partial usage factors for each stress range.
7) Calculate Cumulative Fatigue Usage Factor (CFUF) and determine how many fuel cycles the sleeved tube can operate before reaching cumulative usage of 1.0 and requiring eddy current inspection.

The cumulative usage factor for Palisades [ ] for the analyzed transients in Reference [3] representing an assumed [ ] years of operation after sleeve installation. Based on a usage factor of [ ] a usage factor of [ ] of operation. The sleeved tube would need to be inspected at least [ ] after installation to identify possible fatigue cracking. Since the Palisades Steam Generator Program requires inspection of the tubing at every refueling outage, the sleeve will be inspected prior to observing potential for fatigue cracks.

In addition, it is concluded that the axial load of [ ] Reference [17] used during the qualification leak testing of the non-severed free tube condition, is bounding the axial load of [ ] calculated and used in this document.

Sleeve Design Analysis Conclusion

Based on the analytical evaluation in Section 7.1, the TSP sleeve meets the ASME Section III stress limits and fatigue usage limits and are acceptable for Palisades Nuclear Power Plant. The Framatome TSP Sleeve meets the design requirement of maintaining structural adequacy during normal operating and accident conditions.

7.2 Fatigue Test Loadings

ASME Code, Section XI, IWA-4725.2.1(d) specifies the qualification test requirements for hydraulically expanded sleeves:

> *Specimens representing the expanded sleeve attachment to a tube shall be cyclic tested in accordance with Section III, Appendix II. This fatigue test shall demonstrate that the sleeve attachment can withstand the specified design loadings without exceeding the specified design leakage limit.*

The test loads were originally calculated for the design conditions and transients of the Westinghouse [ ] steam generators. The load ranges were calculated using the W [ ] design transients based on the following conservatisms:

- A single peripheral tube [ ] to the support plate at 100% power conditions,
- All other tubes are free to slide at the tube support plate.
- The support plates are attached to the wrapper and the wrapper is connected to the shell at the elevation of the 2nd TSP.

Test load ranges and required cycles are obtained per ASME Section III, Appendix II, and are based on the number of test assemblies and various factors relating the test conditions to the actual operating conditions. The final axial fatigue load testing sequence for the sleeve (in W [ ] SGs) is shown below for both severed and non-severed tubes.

Table 7-2: Fatigue Cycling Test Parameters

[ ]

The [ ] is not applicable to the lattice bar design for the TSPs at Palisades. Thus, the loadings used for the fatigue cycling are much higher than would be applicable for Palisades. For instance, a maximum tensile load of [ ] force was calculated for a [ ] tube while the maximum tensile load that can be generated at Palisades is the pressure end load from a MSLB accident ([ ] force). Also, the large compressive forces used in the fatigue cycling are extremely conservative for use at Palisades since it is not possible to develop compressive loads on tubes that are allowed to move through the TSP locations. Thus, the fatigue cycling performed for the Westinghouse [ ] series SGs easily encompass the fatigue loadings applicable to the mechanical TSP sleeves installed at Palisades.

7.3 Sleeve Plugging Criteria

A calculation [18] was performed to determine limiting flaw sizes and resulting plugging criteria for mechanical tube support plate (TSP) sleeves that are to be installed in 3/4" outer diameter steam generator (SG) tubes at Palisades Nuclear Power Plant. The evaluation is performed to the requirements outlined in the leakage and structural integrity performance criteria (SIPC) as defined in NEI 97-06 SG guidelines. Satisfying these requirements also effectively satisfy the original criteria for assessing degraded tubing defined in Regulatory Guide (RG) 1.121. This section provides the results and conclusions of the degraded tube analysis and 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 original criteria for evaluation of degraded tubes were prepared many years ago in Regulatory Guide 1.121. However, since then an industry wide program was established to produce a comprehensive program for the care, inspection, and maintenance of steam generators, including updated criteria for assessment of tube degradation. The new requirements, which are based on those originally defined in draft RG 1.121, are now contained in Section 2 of NEI 97-06. Details on the SIPC can be found in the EPRI integrity assessment guidelines which also clarify that detailed and comprehensive discussion on the development and history of the SIPC demonstrating that the recommended criteria is consistent with past practices and meets the intent of the original design basis for SG tube integrity (i.e. RG 1.121).

7.3.1 Performance Criteria

7.3.1.1 Structural Integrity Criteria

The structural evaluation of degraded tubes is performed using the criteria outlined/defined in the Structural Integrity Performance Criteria (SIPC) contained in Section 2.1 of NEI 97-06. The criterion states:

> ***"All in-service steam generator 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."***

This statement provides a set of generic and universal criteria that applies to all steam generator designs. As shown, this new SIPC statement includes a specific requirement for collapse. The criterion for plastic collapse is not associated with implosion due to external pressure, but gross structural collapse associated with contributing loads (like bending). For clarification, a definition of collapse was agreed to by industry representatives and the NRC and the definition was included in the NEI 97-06 documentation. The collapse definition reads:

> ***“Collapse - For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.”***

Based on results from an extensive EPRI testing program of both U-bend and straight tube sections, it is determined that “collapse” of this type is not a credible failure mechanism. This conclusion has been incorporated into the Steam Generator Integrity Assessment Guidelines. With the essential elimination of the collapse criterion, the remaining SIPC criteria that are required for the structural integrity evaluation of degraded tubes include:

- $3\Delta P_{NOP}$ < tube burst – This criterion defines that a safety factor of 3.0 must exist between the normal 100% power operating primary-to-secondary ΔP and the burst pressure of the tube.
- $1.4\Delta P_{ACCIDENT}$ < tube burst --- This criterion defines that a safety factor of 1.4 must exist between the maximum accident (Level C or D) primary-to-secondary ΔP and the burst pressure of the tube.
- $(1.2\Delta P + 1.2\text{Primary Loads} + 1.0\text{Axial Secondary Loads})_{ACCIDENT}$ < tube burst – This criterion basically defines that the combination of pressure loading with other loads that can affect the burst pressure of the tube. The primary loads, including pressure and primary membrane and bending loads shall be evaluated with a safety factor of 1.2 and the axial secondary loads shall be included with a safety factor of 1.0. This criterion addresses the combination of loads such as bending in upper span created by cross flow loading during a MSLB, dynamic loads associated with an earthquake or pipe break, or axial loads associated with differences in temperature between the tubes and shell.

These three criteria are used with the tube loads from the design specification along with the appropriate flaw evaluation methods from the EPRI Flaw Handbook to establish the tube structural limits.

7.3.1.2 Accident Leakage Criteria

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. The criterion states:

> ***“The primary to secondary accident induced leakage rate for any design basis accidents, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1 gpm per steam generator, except for specific types of degradation at specific locations when implementing specific alternate repair criteria as documented in the Steam Generator Program technical specifications”***

Part through-wall flaws have the potential to mechanically tear the remaining ligaments under accident conditions, thus creating a leakage path, referred to as “pop-through”. For volumetric flaws that are predominantly axial in characterization with limited circumferential extent (i.e., <135°), such as the flaws evaluated in this document, the onset of “pop-through” and burst is coincident. Therefore, flaws satisfying the structural performance criteria also satisfy the accident leakage criteria.

The Palisades Plant is subject to a more restrictive Accident Induced Leakage Performance Criteria (AILPC) of **[ ]** per steam generator based on plant Technical Specification. Furthermore, any leakage associated with the C* (C-star) shall also be included in the accident leakage evaluation.

7.3.1.3 Operational Leakage Criteria

The operational leakage integrity of degraded tubes is assessed using the criteria outlined/defined in the performance criteria contained in Section 2.3 of NEI 97-06 [1]. The criterion states:

> ***“The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day.”***

Much like accident leakage criteria, for volumetric flaws that are predominantly axial in characterization with limited circumferential extent (i.e., <135°), the onset of “pop-through” and burst is coincident. Therefore, flaws satisfying the structural performance criteria also satisfy the operational leakage criteria.

Palisades has implemented an additional administrative operational leakage limit of 0.05 gpm (72 gpd).

7.3.2 ANALYTICAL METHODOLOGY

The evaluation of structural and leakage integrity begins with determination of the limiting SIPC for postulated degradation at all locations of interest. This together with sleeve tubing tensile properties allow for the calculation of the pertinent structural limits for each of the applicable flaw models [19] using industry standard software developed to specifically address SIPC criteria.

7.3.2.1 Postulated Degradation

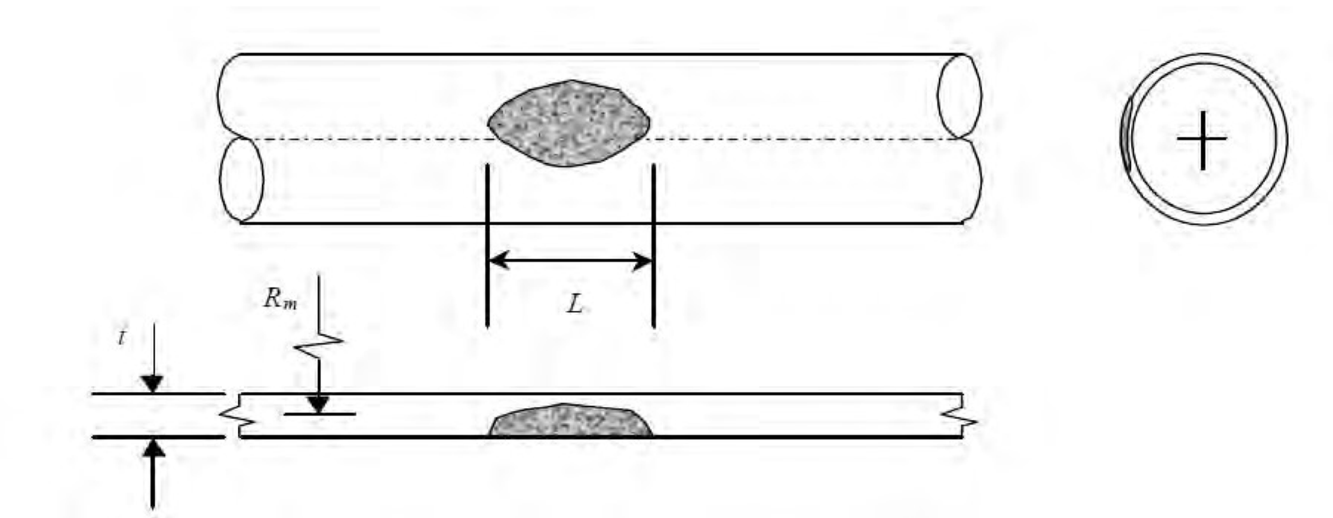
While the Palisades SG tubes are Alloy 600 material and are therefore susceptible to both SCC and volumetric wear, the TSP sleeve tubing material is Alloy 690 material. Of the over 80 units that have SGs with Alloy 690 tubing in operation since 1997 the only tube degradation identified has been volumetric wear. Additionally, all of these plants operate with a primary side temperature (Thot) greater than the 583°F for Palisades. Since SCC is driven by higher temperatures and chemistry environment, and there is over 25 years without SCC identified in alloy 690 tubing, there is reasonable assurance that an Alloy 690 TSP sleeve installed in the Palisades SGs operating under modern chemistry controls will not be susceptible to SCC in the event of a sever in the parent tube at the repaired location.

Therefore, this evaluation will only consider **[** **]** Since the sleeve will only be installed at “egg-crate” or TSP elevations, then the **[** **]** would be caused by wearing on the “egg-crate” support structures.

7.3.2.2 Degradation Model

The EPRI Flaw Handbook identifies the burst equations for each of the potential degradation mechanisms. For tube support wear the evaluation in this document will use the flaw model described in the Flaw Handbook as “axial part-through wall degradation < 135° in circumferential extent”. Since the maximum circumferential extent of a single 100%TW wear scar formed by a long flat bar positioned tangentially to the sleeve tube surface is ~55°, the < 135° model remains bounding. Even for two wear flaws in the same axial plane, the total circumferential extent would be well below the 135° limit established by this model; and this assumes no circumferential separation between the flaws. In reality multiple support flaws occurring in the same axial plane would be adequately separated circumferentially to enable them to be evaluated independently. Hence, this flaw model is appropriate for the evaluation of support wear. A visual representation of this type of flaw is shown in Figure 7-1.

Figure 7-1: Representation of < 135° Flaw Model



7.3.2.3 Structural Limits

There are three types of structural limits considered in an NEI 97-06 analysis which are defined below.

- **Best Estimate Limit (BE)** – The term Best Estimate Structural Limit refers to the actual physical dimensions of tubing degradation such that the limiting SIPC is just met on a best estimate basis. Average tensile properties and nominal degraded tube strength equations are used. No uncertainty terms are included
- **High Probability Structural Limit (SL)** – This term refers to the actual physical dimensions of tubing degradation such that the limiting SIPC is met with 0.95 probability at 50% confidence. Uncertainties in tensile properties and the appropriate degraded tube strength equation are considered.
- **Condition Monitoring Limit (CM)**– The Condition Monitoring Limit refers to measured non-destructive examination (NDE) degradation dimensions such as length, depth or Percent Degraded Area (PDA) such that the limiting SIPC is met with a probability of 0.95 at 50% confidence when NDE sizing uncertainties, tensile property uncertainties and degraded tube strength equation uncertainties are included.

Table 7-3: **Fatigue Cycling Test Parameters** provides a summary of all the uncertainties included in the calculations with each structural limit.

Table 7-3: Fatigue Cycling Test Parameters

[ ]

7.3.2.4 Calculation Process

The EPRI Flaw Handbook describes methodologies which may be used to evaluate the structural integrity of degraded SG tubes. The methodologies provide the means to estimate the strength of the degraded tubing while conservatively accounting for the various uncertainties inherent in the tube integrity process. Framatome has developed general purpose software, as licensed by EPRI, to perform the necessary Flaw Handbook calculations. The program, written in Microsoft Excel Visual Basic Applications (VBA) and referred to as the Flaw Handbook Calculator (FHC), has been validated and verified. The program has been used in recent years to perform inspection specific CMOA evaluations for both RSG and OTSG plants, as well as for the NEI 97-06 degraded tube evaluations for other replacement steam generators.

The FHC employs Monte Carlo techniques to determine the structural limits identified in Table 7-3 while accounting for the various uncertainties associated with each limit. The "best estimate" structural limit, since it is based on average or mean properties, does not account for uncertainties. The "high probability" structural limit accounts for uncertainties associated with burst equation variability (variability between actual integrity test data and the equation, or relation, used to model the data) and material strength variability. Uncertainty parameters are calculated at a required 50% confidence.

The FHC allows the user to specify the level of conservatism reflected in the results by specifying the probability of interest. The results produced by the calculations are such that the actual burst pressure (or axial strength) will have the specified probability of being higher than the calculated result. For example, if the user specifies a probability of 0.95 (95%), and the calculation generates a result [ ]

7.3.2.5 Leakage Integrity

Unless highly unusual circumstances are encountered, such as rapid growth loose parts wear, volumetric flaws do not challenge leakage integrity. Flaw depth must be on the [ ] or more to lead to leakage at NOP or Limiting Accident conditions. If such depths are encountered, smaller volumetric flaws can lead to large leakage events but not tube burst. Burst before leak is expected for large volumetric flaws. Any volumetric flaw that meets structural integrity at 3ΔP or axial load structural integrity should also meet leakage integrity at NOP and accident conditions.

7.3.2.6 Plugging Limit

To determine the acceptable plugging limit for volumetric wear flaws in the TSP sleeve, the worst-case flaw having an axial length of [ ] is reviewed which has a maximum depth [ ] when accounting for NDE sizing uncertainty (i.e., CM limit). For the Palisades SGs, “egg-crate” wear overall has minimal or stagnant growth on average with maximum growth rates of [ ] [20]. Assuming a growth rate of [ ] bounding the maximum observed growth rates, over a single inspection interval of 1.5 EFPY (Palisades is limited to single cycles due to having Alloy 600 mill annealed tubing) provides a margin of [ ] to the technical specification required plugging of ≥40%TW. Therefore, the plugging limit of the TSP sleeve remains the same as any Palisades SG tube of ≥40%TW.

7.3.3 Sleeve Plugging Criteria Conclusion

The results of this evaluation show that flaws [ ] satisfy all performance criteria and allowing for NDE uncertainty and flaw growth, the Palisades technical specification tube plugging limit of ≥40%TW is also applicable to the TSP sleeve.

The Palisades Steam Generator Program currently requires plugging of crack-like indications on detection in the parent tubing, except in areas within the tubesheet covered by the existing alternate repair criteria. This same plug-on-detection criteria applies to the parent tubing portion of the pressure boundary in the sleeve joint regions. SG tube surveillance requirements continue to ensure that defective tubes will be repaired or removed from service upon detection.

7.4 Thermal-Hydraulic Analysis

The effect of TSP sleeve installation on the RSG performance was analyzed for heat transfer, flow restriction, and steam generation capacity [ ] [21]. Depending on the sleeve length, number of plugs installed, and number of sleeves installed, the sleeve to plug ratio will vary between roughly [ ] Table 7-4 summarizes the effects on the RCS from sleeving and plugging of tubes noting that the sleeve to plug ratio is a minimum value which increases with the number of sleeves installed.

Table 7-4 : Thermal-Hydraulics Summary

[ ]

At the conclusion of each SG ECT inspection, the effective plugged percentage of each S/G is calculated and reported in the Operational Assessment for the next operating cycle. Comparison to the allowed plugging percentage will demonstrate proper operation of the steam generator with respect to heat transfer for the next cycle.

7.5 Sleeved Tube Seismic Considerations

Palisades is in an area of very low seismic activity. Therefore, the original postulated OBE loads are much greater than the Palisades design basis. Ground accelerations for the operational basis earthquake used for containment design purposes and all seismic Class I structures are 0.10g applied horizontally and 0.07g applied vertically. In addition, ground accelerations of 0.2g horizontal and 0.13g vertical are used for the design basis earthquake [22].

The recommended surface acceleration value was 0.05 g; however, [ ] was used for the Facility design earthquake (OBE) and [ ] as the hypothetical earthquake (SSE) [23].

The effect of TSP sleeve installation on the stresses due to seismic loading, as well as other concurrent design basis loads in the tubes, is evaluated and shown to be acceptable in Reference [13]. That evaluation does not consider the impact of the increased weight on the tube support loading and deformation. The impact on loading of the tube support is based on the change in the supported weight and demonstrates that the change is negligible.

The Operating Basis Earthquake (OBE) loads were calculated using the finite element model subjected to [ ] of horizontal acceleration. The OBE loads were determined at 70°F since this results in the maximum density of water. For consistency with the worst-case sleeve loads [ ] tube condition was used to determine the additional tubesheet sleeve load due to OBE seismic. The [ ] case was also used for the non-leaking TSP sleeve load due to OBE since it resulted in conservatively high loads (since the tube at the TSP carried no load in the [ ]). The density of the tube was modified to include the weight of the water in the tube, the weight of the tube, and the weight of the water outside the tube that influences the tube vibration (added mass). The added mass was assumed to equal the mass of the displaced fluid [13]. The OBE was assumed to occur at times other than at the maximum transient stress. Therefore, the [ ] used in the fatigue analyses and were neglected in the fatigue analyses based on Assumption 7 of Reference [13].

The added weight of a single sleeve, with a maximum [ ] The wet weight of a single steam generator tube, with span length of 38.75 inches, equal to the distance between adjacent tube supports, is [ ] lb. The percent change in weight with the maximum length sleeve added is approximately [ ] for a single tube. The natural frequency of the tube is not impacted as the additional weight is fully supported by the tube support. For seismic loading with no change in natural frequency, increased weight translates directly to increased loading at the support.

The Palisades steam generator is designed with 8219 tubes, which corresponds to 16,438 tube spans due to the U-tube design. An increase in seismic loading of no more than [ ] is considered negligible. Therefore, a total increase in weight of no more than [ ] in the tube span is considered acceptable. Based on this evaluation, it is acceptable for as many as 2740 sleeves to be installed (16438 tubes x [ ]) at a given tube support.

7.6 Flow Induced Vibration

The flow-induced vibration (FIV) analyses [16] evaluated fluid elastic stability margins (FSM) and random turbulence vibration responses for TSP sleeves installed at all TSPs on the hot leg. The natural frequencies and mode shapes of the RSG tube and sleeve combination were determined.

The FIV tube model included the hot leg, U-bend, and cold leg tubing from tubesheet to tubesheet. This model allowed cross flow loads to be applied to the U-bend tubing for evaluating the tube support plate sleeves. The model included [ ] The model was individually evaluated for a sleeve installed at each of the eight hot leg tube support locations. An un-sleeved (virgin) tube was also modeled.

For FSM, results were compared to either the Framatome acceptance criteria [ ] or to the virgin tube results. If the results showed that a given [ ]

For the random turbulence response, tube displacement results were compared to the virgin tube displacement behavior. The virgin tubes have not exhibited tube-to-tube wear after 32 years of operation. This indicates that the virgin tube displacements due to random turbulence have not been sufficient to cause tube-to-tube contact and wear during operation. If the results showed that a given tube displacement was equal to or less than the result for the virgin tube, the result was considered acceptable.

The FIV analyses [16] indicate that the sleeves are acceptable for installation based on FSM and random turbulence vibration considerations.

7.7 Design Analysis Conclusion

Based on the Palisades analytical evaluations contained in Section 7.0, it is concluded that the TSP sleeves described in this document meet all pertinent design requirements of Section 4.0 with margins. The analytical evaluations documented the ASME Section III stress analysis, seismic, flow induced vibration, fatigue usage, fatigue loadings, Regulatory Guide 1.121 criteria and thermal-hydraulic impacts. Therefore, the sleeves are acceptable for installation at the Palisades Nuclear Plant (PNP).

The Framatome Alloy 690 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 current steam generator (SG) tubing. The applied stresses and fatigue usage for the repaired sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of 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.

The Alloy 690 repair sleeve depth-based structural limit was determined using the RG 1.121 guidance and the pressure stress equation of ASME Code, Section III. The current PNP Administrative Controls Technical Specification (TS) 5.5.8c, Steam Generator (SG) Program, limit of a depth equal to or exceeding 40% of the nominal tube wall thickness is conservative. A sleeved tube is plugged on detection of degradation in the sleeve/tube assembly.

Evaluation of the repaired SG tube by 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. The sleeve is fabricated from Alloy 690 which provides enhanced corrosion resistance compared to the current SG tubing.

The implementation of the proposed sleeve 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 SG tube rupture (SGTR) analysis described in the PNP Updated Final Safety Analysis Report, (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 would result in lower total primary fluid mass release to the secondary system. The minimal leakage that could occur from the repaired sleeve/tube assembly during plant operation is well within the TS leakage limits. Therefore, the proposed sleeve does not involve a significant increase in the probability or consequences of an accident previously evaluated.

8.0 ASME SECTION XI MECHANICAL TESTING OF THE SLEEVE

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. The mechanical testing and the results are discussed below.

8.1 Specimen Description

The mechanical test specimens consisted of a total of [ ] These specimens had three tube conditions: [ ] Note that all the samples shown in Table 8-1 were used for the correlation between the [ ] value and the tube [ ] The [ ] samples listed in Table 8-1 were used for the leak rate analysis.

Figure 8-1: ¾" TSP Sleeve Test Specimens

Table 8-1: 0.750 in TSP Qualification Specimen Assembly

The specimen tube conditions simulated the following conditions:

The specimens that were fully [ ] between the outermost and the second ribbed expansions, on each end of the specimen, were tested to assure that a single joint could carry the operating and accident loads.

1) Since this condition is not expected to occur in the steam generator, these samples were used for expansion sizing [ ]
2) The [ ] samples simulated a leaking tube condition since a [ ] was placed in the center of the tube. These specimens represent the expected worst case condition within the steam generator, i.e., sleeving a tube which is leaking through a defect at the TSP.

The tubing used for the qualification specimens had a yield strength ranging from [ ] heat treatment was performed to purposely lower the tube yield strength. This treatment also produced a heavier oxide.

During fabrication the sleeves were stress relieved and oxidized using a temperature of [ ] to relieve any stresses from the machining /fabrication process.

All sleeves were installed using the Framatome hydraulic expansion system as described in Section 9.0, with eight simultaneous expansions made in the sleeve. A range of tube OD expansion sizes were produced during testing to simulate possible steam generator 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 ribs on the OD of the sleeve to better engage the tube [ ]

8.2 Mechanical Test Description

Table 8-2 provides a summary of the tests performed on each specimen. These tests are described in more detail below. Tests were performed in the order listed.

Table 8-2: 0.750 in TSP Qualification Specimen Test Sequence

Table 8-3 shows the test parameter used on the samples. Each Test will also be described in detail in the following sections.

Table 8-3: Leak Test Parameters

[ ]

8.2.1 Primary Side Room Temperature Leak Test

All specimens received either an initial room temperature leak test or a no-load room temperature leak test. In some cases, the no load room temperature leak testing was performed after the initial hot leak tests. Refer to Table 8-2 for the test sequence for each specimen. Table 8-3 contains the leak test parameters.

Room temperature leak tests were performed by connecting a high-pressure pump to one end of the specimen while capping off the opposite end. The pump was used to establish an initial pressure in the system. After reaching the approximate test pressure, the pump was isolated, and a pressure generator was used to achieve and maintain the required test pressures. The fine graduations of the pressure generator were used to add a known volume of water into the system, over a period, to maintain test pressures.

[ ] then it was necessary to use a pump to maintain system pressure while the volume of water was collected **[ ]** The test setup for leak testing is shown schematically on Figure 8-2.

Figure 8-2: Leak and Test Setup

[ ]

Room temperature leak testing was performed at a minimum pressure of [ ] for a minimum of [ ] are representative of conditions expected within the Palisades steam generators. Typically, the leak rates for these specimens remained relatively unchanged as testing progressed.

8.2.2 Operating Temperature Leak Test

Operating temperature leak testing was performed on various specimens during the qualification process, as shown in Table 8-2.

Prior to starting hot leak testing on the specimens, thermocouples were welded on the tube OD above the uppermost expansion, between the upper and lower expansions, and below the lowermost expansions. A thermocouple was also placed inside the specimen to measure the water temperature. These temperatures were recorded during the testing. The specimens were mounted on the operating temperature leak test fixture, as shown on Figure 8-3.

Figure 8-3: Operating Temperature Leak Test Setup

[ ]

The system was initially filled with water. Power was applied to heat tape wrapped around a tube below the specimen to increase the water temperature. The temperature was increased [ ] was reached, while maintaining the minimum pressure of [ ] If leakage occurred, it was captured by the condensation pipe connected to the side of the test specimen and directed into the [ ] The top of the [ ] was blocked so that vapor could not escape during testing.

After reaching the test temperature, the test pressure was held for a minimum of 30 minutes while any leakage was collected. Each specimen was initially insulated to keep the difference in temperature between the water and the tube, [ ] as close as possible to simulate locked tube loads. The insulation was then removed, and the leak test was performed again with a higher differential temperature. After the testing was complete the specimen was allowed to cool naturally.

[ ] samples prior to cycling and [ ] tests were performed after cycling. During testing, the typical temperature differential between the sleeved tube and the internal primary side water was [ ] for the insulated tests and [ ] for the uninsulated tests. The uninsulated test differential is similar as that expected in the steam generator, or approximately [ ]

8.2.3 Cyclic Testing

The adequacy of the sleeve and its joints to withstand operational loading was demonstrated by subjecting the 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 steam generator to determine if the plant operating conditions would degrade the leakage performance of the sleeve joint.

The specimens which were cyclic tested are listed in Table 8-2.

Table 8-4 provides the parameters for the cyclic testing.

Table 8-4: Cyclic Test Parameters

[ ]

Axial Cycling

Axial cycling was performed on the [ ] specimens by fixing each end of the test assembly into a cyclic testing machine. The specimens were then axially cycled to the conditions shown in Table 8-4. This testing represents 10 years of structural design life which qualifies the TSP sleeve for the remainder of the currently licensed plant life at Palisades.

The loads developed for testing were based on tubes locked at the support plates. Locked tube loads result due to the differential thermal growth of the tube, sleeve, shell, and wrapper. These loads, in the steam generator, are displacement limited. Due to equipment limitations, axial cycling was conservatively performed as a load limited test. The development of the loads was per Reference [24].

The tubes in the Palisades steam generator are not locked since they have a lattice (egg crate) design which has a looser fit than a drilled hole. Therefore, the axial loads tested are expected to be much higher than for the Palisades condition.

None of the samples recorded sleeve joint motion during testing.

Thermal Cycling

Thermal cycling was performed on the [ ] Reference [25] outlines the development of the test temperatures. The heat-up and cooldown rates for each cycle were nominally [ ] The total hold time during the [ ] was a minimum of [ ] hours. Additional thermal cycles were also performed on some of the specimens during operating temperature leak testing.

Thermal cycling is performed to determine if the differences in the thermal expansion coefficient for the sleeve and tube will affect the hydraulic expansion joints. The susceptibility of the hydraulic joints to thermally relax when exposed to plant operating temperatures is also tested. The [ ] minimum hold time was set to allow the mechanical joints to be at temperature long enough for thermal relaxation to occur. Based on data for alloy [ ] some thermal relaxation may occur within the first couple of hours of achieving temperature. It is not expected that thermal relaxation will occur at temperatures as low as [ ] however.

None of the samples recorded sleeve joint motion during testing.

Pressure Cycling

Pressure cycling was performed on the [ ] specimens by pressurizing the inside of the sample between [ ] psig minimum for at least [ ] cycles, as developed in Reference [26]. This test represents the startup and shutdown pressure cycling to which a sleeve is exposed during 10 years of design life.

No sleeve-to-tube movement was noted during testing.

8.2.4 MSLB Leak Test

After the leak and cyclic tests were complete, the sample pressure was increased to a minimum of 2780 psig with a hold time of [ ] minutes minimum to represent the maximum pressure expected from a main steam line break accident. This pressure is shown in Table 8-3 and the method of testing is the same as described in Section 8.2.1. The leak rates measured during this testing, performed at room temperature, were higher than those previously recorded during room temperature leak testing. No joint failure was noted during testing.

8.2.5 Leakage

After qualification testing was complete on the ¾" mechanical TSP sleeve samples, data analysis [27] was performed to determine the expected leak rates.

Leak rate values are given for both beginning and end of life leakage, with beginning of life defined as the leakage prior to cyclic loading and end of life after the 10-year cyclic loading. Only the non-severed specimens, [ ] are discussed in this section since this is the condition expected in the Palisades steam generators.

To determine the expected steam generator leak rates, the following steps were performed to analyze the qualification testing data.

- The sleeve installation data was reviewed to arrive at a distribution of expected expansion sizes, in low, medium, and high yield tubes, to a 95% confidence level.

- The percentage of [ ] The distribution of tube yield strength was determined to be [ ] for Palisades. A graph of the actual plant data is shown in Figure 8-4.

Figure 8-4: Distribution of Tubing Yield Strength

[ ]

- The leak rates measured during each individual test were averaged by [ ] and tube yield strength [ ] distribution, resulting in a 3 x 3 matrix. The percentage of occurrences for each of the average leak rates within this matrix were determined so that a final average leak rate for each test could be determined. The final leak values for each test are shown on Table 8-5. Each test is listed individually so that the effects of the test parameters on the leak rate can be noted. All leak values are reported in gallons per day.

Table 8-5: Average Leak Rates

[ ]

The [ ] rate data was used to determine the final steam generator average value.

The method used to determine average leakage was also applied to the data for the specimens with [ ] between the hydraulic expansion joints. The final leak rates were determined for non-locked tubes at the normal operating and main steam line break pressure conditions for the Palisades RSG's. These leak rates are based on a target tube [ ]

The operational leak rate is monitored through plant equipment and does not limit the total number of sleeves that can be installed. The MSLB leakage value does limit the total number of sleeves that can be installed. For the MSLB leakage calculations in the Steam Generator Operational Assessment, every installed sleeve is assumed to be leaking at the sleeve MSLB leakage rate [ ] gallons per day per sleeve. The Palisades plant has a total MSLB accident-Induced leakage [ ] per steam generator. The C* alternate repair criteria assigns an MSLB leakage of [ ] gallons per day per channel head. If we also assign an MSLB leakage of [ ] gpd to sleeve leakage, approximately [ ] sleeves could be installed per steam generator.

The Palisades Technical Specification Operational and Accident-Induced leakage limits assure that any postulated leakage in the limiting SG will be less than the bounding faulted condition leakage necessary to ensure that offsite doses remain a small fraction of the 10 CFR Part 100 reactor site criteria and that control room doses remain within 10 CFR 50 Appendix A, GDC 19 limits. The operational and accident-induced leakage contribution from the installed sleeves is held within the Palisades Technical Specification Operational and Accident-Induced leakage limits.

8.3 ASME Section XI Testing Conclusions

Mechanical tests were performed on sleeved mock-up tubes to determine the leak rate properties. These tests included initial leak rates, axial, thermal, and pressure cycling, and final leak rates at normal operating and main steam line break conditions.

The results of the mechanical testing of the sleeve samples demonstrates they provide an adequate safety factor for normal operating and postulated accident conditions. Mechanical testing determined the installed sleeve would also withstand the cyclical loading resulting from power changes in the plant and other transients.

9.0 SLEEVE INSTALLATION

The installation of the hydraulically expanded tube support plate sleeve is accomplished remotely, using tooling attached to a steam generator manipulator.

The sleeve installation and inspection methods minimize the personnel radiation exposures in accordance with ALARA principles. The operations required for the mechanical TSP sleeve installation are:

- Pre-installation tube eddy current inspection.
- [ ]
- Sleeve insertion and hydraulic expansion of rib areas above and below the TSP.
- Partial hydraulic expansion of the center of the sleeve.
- Post-installation sleeve/tube eddy current inspection.

9.1 Pre-Installation Tube Eddy Current Inspection

An eddy current test (ECT) of the tubes to be sleeved is performed to verify there is no degradation of the tube in the upper and lower expansions. If not already known, the distance from the TSP to the diagonal supports can also be confirmed by eddy current testing to assure the sleeve can be installed in locations near the diagonal supports. Lastly, any observed dents/dings in the target tube are noted for further testing.

[

]

Since the TSP sleeve is inserted deep into the tube, an optional go-gaging step may be performed to ensure that any small tube dents at the support plate intersections (detected by ECT) will not restrict the sleeve. The go-gage, with similar dimensions as the sleeve, would be mounted on a flexible sheath and would be inserted to the depth of the sleeve. If the go-gage cannot pass through a restriction, then the tube will not be sleeved. The pre-installation eddy current tests and successful go-gaging confirm that the tube can be repaired by sleeving.

9.2 Sleeve Insertion and Hydraulic Expansion

The following steps are used to install a sleeve into a steam generator tube. These steps are performed in batches to improve productivity.

1. The first part of the installation process is to assemble and calibrate the hydraulic expansion system, shown in Figure 9-1. The pressure and piston transducers in the expansion box are calibrated.
2. The eight-bladder hydraulic expander is placed on the end of the expander insertion sheath. The expander is inserted into a solid test block and pressurized to the maximum installation pressure to perform the final calibration of the expansion system.
3. A sleeve is placed over the hydraulic expander tool. The eight bladders on the expander tool are pressurized to [ ] to hold the sleeve onto the tool during insertion into the tube.

4. The expander tool and sleeve are loaded into an actuator and inserted into the steam generator tube to the elevation of the target TSP. A positioning mark or hard stop on the hydraulic expander insertion sheath insures proper positioning at the TSP. See Figure 9-2 for a representation of the sleeve position at the TSP.
5. The expansion of the sleeve into the tube is performed by pressurizing the eight bladders [

   ] The point at which the sleeve contacts the tube is identified by the change in slope on the pressure vs. displacement curve (Figure 9-3). [

   ]
6. After the process is complete the expansion system automatically depressurizes, process data is automatically saved, and the expansion tool is removed from the sleeve / tube and prepared for the next sleeve.
7. After the eight expansions are made, a partial, pressure controlled single bladder expansion is made in the center of the sleeve. This expansion is used to preload the four upper and four lower expansion joints. [

   ]

9.3 Post-Installation Eddy Current Inspection

After the sleeves are installed, an eddy current examination is performed to verify the location of the sleeve relative to the defect region and obtain a baseline inspection of the sleeve and tube. This will allow future comparisons of the sleeve and tube conditions.

9.4 Plugging of a Defective Tube

If a sleeved steam generator tube is found to have an unacceptable defect in the pressure boundary portion of the tube or sleeve, the steam generator tube can be taken out of service by installing standard, site approved mechanical plugs and stabilizers.

Figure 9-1: Expansion System Schematic

Figure 9-2: Sleeve Installation

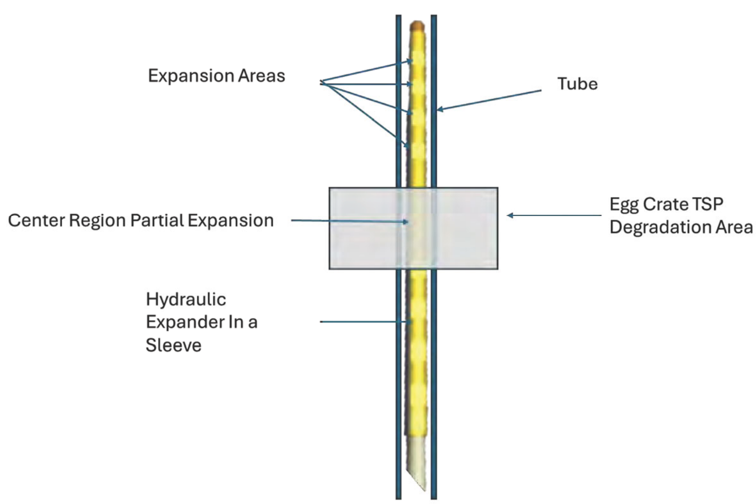


Figure 9-3: Typical Process Expansion Curve

9.5 ALARA CONSIDERATIONS

9.5.1 Personnel Exposure

The steam generator repair operation is designed to minimize personnel exposure during installation of sleeves. To achieve this, the manipulator is installed from the manway without entering the steam generator and is operated remotely from a control station outside the containment building. The sleeve delivery system allows the sleeve to be positioned on the expansion tool from outside the steam generator. The expansion tool and sleeve are then delivered into the steam generator tube remotely.
If tooling repair is needed, then repairs are performed in an area which does not result in significant radiation exposure. Air, water, and electrical supply lines for the manipulator and actuator tool are designed so they do not become entangled during operation. The operations required to install a sleeve is like the installation of a plug in a steam generator tube. In summary, the sleeve installation process is designed to minimize personnel exposure and is expected to have the same impact as steam generator plugging evolutions commonly conducted at nuclear power plants.

9.5.2 Radioactive Waste

The mechanical sleeving process generates a minimal amount of radioactive waste. The components which encounter the tube ID, and are the most highly contaminated, are considered disposable. These are the urethane bladders, stainless steel hydraulic expanders, and nylon expansion sheaths. A minimal amount of wet waste is generated during bladder change out and this can be expected to be contained and readily absorbed by rags or other decontamination materials.

Additional radioactive waste is produced by disposable protective clothing, tape, plastic bags, and other material used during normal operation in a contaminated environment. This waste is not addressed in this document and is processed through the Palisades facility designated for handling this material.

10.0 SLEEVE EXAMINATION

10.1 Eddy Current Requirements for Tubing with Sleeves

Prior to sleeve installation, precise measurements of the location(s) of the defect must occur. It is imperative the parent tubing be free of any defect in the location of the sleeve expansions. The precise measurements of all stress corrosion cracking (SCC) were line-by-line sized from the last steam generator inspection in 2024. All results from the line-by-line sizing are loaded into the appropriate database for ease of access.

An eddy current examination is performed on both the sleeve and parent tubing behind the newly installed sleeve following sleeve installation. Utilizing the bobbin coil, this inspection technique is utilized to verify proper sleeve position axially in the tube. Utilizing the +point probe, a baseline inspection of the new primary pressure boundary is acquired and analyzed. This baseline data is part of the permanent record for the steam generator and available when performing future in service inspections of the sleeves to use as a reference.

For future in service inspections, eddy current inspections on the full length of the sleeve are added to the inspection program of the steam generators. This inspection utilizes the +point probe to ensure appropriate coverage is maintained in the sleeve and when transitioning from the sleeve to the parent tubing. In tubing where only one sleeve is installed, the inspection will cover the entire length of the sleeve to include a minimum of two inches below the lower sleeve end into the parent tubing, and a minimum of two inches above the upper sleeve end into the parent tubing. In tubing where multiple sleeves exist, the same method applies but the parent tubing between the sleeves are inspected full length with the +point probe. The minimum inspection of two inches outside of the sleeve ends can be extended if necessary.

10.2 Eddy Current Qualification

Currently, sleeve samples that include parent tubing, are being fabricated to perform an eddy current technique qualification. The technique will verify the proper sleeve position and provide eddy current techniques to detect and size degradation that may appear. Requirements set forth in the current Rev. 8 Electric Power Research Institute (EPRI) Guidelines, Appendix H outlines how to perform a performance demonstration for eddy current examination techniques.

A. Detection data set:

   a. Minimum detection data set is 11 flawed grading units
   b. Recommended the data set be uniformly distributed over the depth range of [ ] to 100% through wall (TW). It is acceptable to use a lower percent TW criterion for detection rather than the [ ] value.
   c. Technique Acceptance Criterion of probability of detection (POD) [ ] confidence level (CL).

B. Sizing data set:

   a. Minimum sizing data set is 16 flawed grading units
   b. Recommended the sizing data set include the detection data set

c. Recommended the remainder of the sizing data set have maximum measured depths less than [ ] of the nominal tube wall thickness. It is recommended the data set be uniformly distributed over the depth range of 20% TW to [ ] TW, as is reasonably achievable.

To ensure these requirements are met, a total of [ ] qualification samples which includes a variety of defects in multiple locations are being fabricated. [ ] The table below shows targeted flaw type, location, and total flaws.

Table 10-1: Flaw Summary

[ ]

Once the sample fabrication is complete, post flaw acquisition and analysis will occur. The results of the eddy current techniques are documented in the Eddy Current Qualification Site Report. A total of nine detection and sizing techniques are planned:

A. Axial ODSCC at the parent tube OD
B. Axial ODSCC at the sleeve OD
C. Axial PWSCC at the parent tube ID
D. Axial PWSCC at the sleeve ID
E. Circumferential ODSCC at the parent tube OD
F. Circumferential ODSCC at the sleeve OD
G. Circumferential PWSCC at the parent tube ID
H. Circumferential PWSCC at the sleeve ID
I. Tapered wear in the parent tubing OD with a Technical Justification to confirm wear at the sleeve OD is detectable if the parent tube is not present.

Projected schedule is to have all samples fabricated to start the qualification in March 2025 and have the qualification report complete by the end of May 2025 prior to TSP sleeve installation in the Palisades Steam Generators.

10.3 Eddy Current Sleeve Experience

The inspection techniques currently in development have been studied and documented in the past for the same design of sleeve and same size tubing that is present in the Palisades steam generators and in Once Through Steam Generators (OTSG) sized tubing.

10.3.1 Mechanical Tube Support Plate (TSP) Sleeve Qualification for ¾" OD Tubes

Based on previous eddy current qualifications, being able to detect flaws in the parent and/or sleeve tubing has previously been demonstrated. Framatome [ ] for ¾" OD Tubes [5], established an eddy current technique qualified for a mechanical tube support plate sleeve for 0.750" OD x 0.043" WT tubing. For this qualification the [ ] was used for defect detection. This probe has one coil that is circumferentially wound for axial indications and one coil that is axially wound to detect circumferential indications. The [ ] is like the current +point coil with axial and circumferential wound coils.

The sample utilized in this qualification included [ ] contoured bottom holes, 20% TW, in both the sleeve ID and parent tube OD in different regions of the sample. The regions included the sleeve ID in expanded and non-expanded areas and in the parent tube OD in the expanded and non-expanded areas. All defects in the sample were detectable.

10.3.2 Evaluation of +Point Probe for OTSG Tubing

Although not in ¾" tubing, Framatome performed a +point evaluation for [ ] This is documented in Framatome [ ] [28]. The sleeving process of OTSG tubing is similar to the sleeving process of the ¾" steam generator tubing. The only difference being the difference in tube size. Test specimens were created for this evaluation following the same design as the ones used in technical [ ]

The result of this evaluation is the +point probe successfully met the requirements which was outlined in the ASME section XI code as the capability to detect 20% Flat Bottom Hole (FBH) in the parent tube OD.

10.3.3 Detection of Rolled Sleeve Tube Defects in OTSGs

Another reference document is from [ ] Detection of [ ] [29]. Work performed in this document was on OTSG tubing 0.625" OD x [ ] ID x [ ] This evaluation was performed in accordance with EPRI Guidelines Rev. 5, Appendix H.

A total of [ ] were manufactured for this evaluation. The degradation added to the samples was laboratory induced volumetric intergranular attack (IGA) with mixed mode cracking. A destructive evaluation (DE) was performed after the NDE evaluation to be able to compare eddy current and DE results.

The POD [ ] with a lower confidence level of [ ] for detection of defects with a maximum depth of [ ] and greater for flaws in the parent tube. The POD exceeds [ ] with a lower confidence interval of 90% for detection of defects with a maximum depth of [ ] and greater for flaws in the parent tube. The lowest DE Flaw depth detected by ECT in the parent tube was an axial ODSCC flaw of [ ] The results of this qualification proved that the +point probe exceeds the EPRI Appendix H requirements for flaw detection.

The qualification outlined in this document was later used to develop the EPRI qualified techniques 96912.1 and 96912.2 for Axial and Circ ODSCC in the sleeve and parent tubing with a +point probe following the strict requirements set forth by EPRI which included a peer review to validate the results.

10.3.4 Eddy Current Sleeve Experience Conclusion

The documents listed above confirm the successful detection of mixed mode intergranular attack and stress corrosion cracking in the parent tubing of OTSG rolled sleeves. Detection of this type of degradation was demonstrated on laboratory IGA/SCC samples with [ ] and the EPRI qualified techniques 96912.1 and 96912.2. Earlier qualifications (document 1241690) used EDM notches and drilled holes to demonstrate detection in critical areas of the sleeve. Detection has been demonstrated with 20% TW flat bottom holes and 40% TW EDM axial and circumferential notches in the parent tubing. OE in the industry also demonstrated the successful detection of axial ODSCC (groove IGA) below the sleeve in the parent tubing with the +point probe at OTSG plant.

Based on the information and experience above, Framatome has confidence in meeting and exceeding all requirements to perform a successful ECT qualification.

10.4 Eddy Current Restriction on Sleeve Installations

Installation of sleeves in both the hot and cold leg tube support locations of the same tube will impede the ability to efficiently inspect the un-sleeved section of the tube. The sleeve is intended to only be installed at the hot leg tube support locations to facilitate eddy current inspection of the cold leg and U-bend portions of the tube.

11.0 REFERENCES

References identified with an (*) are maintained within Palisades Records System and are not retrievable from Framatome Records Management. These are acceptable references per Framatome Administrative Procedure 0402-01, Attachment 7.

1. NEI 97-06, Rev.3, “Steam Generator Program Guidelines,” January 2011.
2. ASME Boiler and Pressure Vessel Code, Section II, III, V, and XI.
   - Section II, Materials. 2019 Edition.
   - Section III, Rules for Construction of Nuclear Facility Component. 2019 Edition.
   - Section V, Nondestructive Examination. 2019 Edition.
   - Section XI, Rules for Inservice Inspection of Nuclear Reactor Facility Components. 2007 Edition with Addenda through 2008.
3. [ ]
4. [ ]
5. [ ]
6. EPRI Report 1019044, “Steam Generator Management Program: Improvement Factors for Pressurized Water Reactor Steam Generator Tube Materials,” December 2009.
7. EPRI Report 1024992, “Steam Generator Management Program: Alloy 600 Steam Generator Tube Experience,” June 2012.
8. EPRI Report MRP-258, “Resistance to Primary Water Stress Corrosion Cracking of Alloy 690 in Pressurized Water Reactors,” August 2009.
9. EPRI Report 1003589, “Pressurized Water Reactor Generic Tube Degradation Predictions, U.S. Recirculating Steam Generators Alloy 600 TT and Alloy 690 TT Tubing,” July 2003.
10. M. Lucien Gander, "Design and Fabrication Issues for New and Replacement S.G,” presented at the EPRI Alloy 690 Workshop, April 1989.
11. 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.
12. B.P. Miglen and G.J. Theus, “Stress Corrosion Cracking of Alloy 600 and Alloy 690 in All-Volatile- Treated Water at Elevated Temperatures,” EPRI Report NP-5761M, May 1988.
13. [ ]
14. [ ]

15. [ ]

16. [ ]

17. [ ]

18. [ ]

19. EPRI Report 3002005426, “Steam Generator Management Program: Steam Generator Degradation Specific Management Flaw Handbook, Revision 2,” October 2015.

20. [ ]

21. [ ]

22. *Palisades Final Safety Analysis Report Section 5.7.1, Revision 35.

23. *Palisades Final Safety Analysis Report Section 2.4.4, Revision 35.

24. [ ]

25. [ ]

26. [ ]

27. [ ]

28. [ ]

29. [ ]

Enclosure 6

PNP 2025-003

Framatome Affidavit, Proprietary Information Notice

2 pages follow

A F F I D A V I T

1. My name is Philip A. Opsal. I am Manager, Product Licensing for Framatome Inc. (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Engineering Information Record 51-9385467-002, entitled, "Steam Generator Mechanical TSP Sleeve Qualification Report for ¾" Tubes at Palisades Nuclear Power Plant" referred to herein as "this Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

   (a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6 (c), 6(d), and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

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

Executed on February10, 2025.

Philip A. Opsal

Philip A. Opsal
Manager, Product Licensing, Framatome Inc.