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

August 1, 2023

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 – AUTHORIZATION AND SAFETY EVALUATION FOR RELIEF REQUEST 4RR-10 (EPID L-2023-LLR-0014)

LICENSEE INFORMATION

Recipient's Name and Address:	Mr. Edward Casulli Site Vice President Susquehanna Nuclear, LLC 769 Salem Boulevard NUCSB3 Berwick, PA 18603-0467
Licensee:	Susquehanna Nuclear, LLC
Plant Name(s) and Unit(s):	Susquehanna Steam Electric Station (Susquehanna), Unit 2
Docket No(s).:	50-388

APPLICATION INFORMATION

Submittal Date: April 10, 2023

Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.: ML23100A128

Supplement Date(s): April 11, 2023

Supplement ADAMS Accession No(s).: ML23101A153

Applicable Inservice Inspection (ISI) Program Interval and Interval Start/End Dates: The unit's fourth 10-year ISI interval began on June 1, 2014, and is scheduled to end on May 31, 2024. The fifth 10-year ISI interval is scheduled to begin on June 1, 2024, and is scheduled to end on May 31, 2034.

Alternative Provision: The licensee requested an alternative under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(2) for the duration of one cycle of operation, Cycle 22. Cycle 22 of operation spans the remainder of fourth 10-year ISI interval and the initial portion of the fifth 10-year ISI interval.

ISI Requirement: The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements applicable to this request are in Section XI, IWA-4000 and include:

- IWA-4131.1, which states, in part, that for repair/replacement activities involving the items identified in IWA-4131.1(a), the alternative requirements of IWA-4131.2 may be used.
- IWA-4131.2, which states, in part, that for repair/replacement activities involving items identified in IWA-4131.1, the requirements in IWA-4131.2(a) may be used in lieu of those in IWA-4000.
- IWA-4321(c), which states that threaded joints in which the threads provide the only seal shall not be used in Class 1 piping systems, and that if a seal weld is employed as the sealing medium, the stress analysis of the joint shall include the stresses in the weld resulting from the relative deflections of the mated parts.

Applicable Code Edition and Addenda: The unit's code of record for the fourth 10-year ISI interval is the 2007 Edition and 2008 Addenda of the ASME Code, Section XI. Upon entry into the fifth 10-year ISI interval, the licensee will update its code of record to the 2019 Edition of the ASME Code, Section XI.

Brief Description of the Proposed Alternative: The licensee proposed to leave the existing leakoff port plug threaded connection of the core spray injection-to-reactor vessel valve 252F007B in service for one cycle of operation, Cycle 22, in lieu of replacing the plug with an ASME Code-compliant, seal-welded plug during the spring 2023 refueling outage. Because Cycle 22 of operation will span portions of both the fourth and fifth 10-year ISI intervals, the licensee requested that the proposed alternative covers the remainder of the fourth and the initial portion of the fifth 10-year ISI intervals through the spring 2025 refueling outage.

As background information, the licensee stated in its request that during the planned September 2022 maintenance outage and investigation of the source of ongoing drywell unidentified leakage, the reactor bottom head drain bypass valve 244F103 was found to be leaking from the valve packing leakoff port plug threaded connection. The licensee's root cause analysis determined that the plug was not replaced with an ASME Code-compliant pressure-retaining seal-welded plug during the 1988 valve pack redesign that made the leakoff port plug a pressure-retaining component. In the spring 2023 refueling outage following the September 2022 maintenance outage, the licensee performed the extent-of-condition evaluation and identified 14 additional valves in Unit 2 with leakoff port plug threaded connections that were not seal welded in accordance with the ASME Code, Section II when the valves were redesigned in 1988. The licensee stated that it emergently scheduled all 14 valves to have the ASME Code-compliant repairs installed in the spring 2023 refueling outage. The licensee repaired all noncompliant valves except valve 252F007B, which is the subject of the licensee's relief request. The licensee determined that isolating valve 252F007B to safely perform the ASME Code repair was not possible in the spring 2023 refueling outage because of the valve's proximity to the reactor vessel and size of the line.

The ADAMS Accession Nos. identified above have additional details on Relief Request 4RR-10.

STAFF EVALUATION

The NRC staff evaluated the Relief Request 4RR-10 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on whether compliance with the specified requirements of 10 CFR 50.55a(g) or portions thereof would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

On April 12, 2023 (ML23103A470), the NRC verbally authorized the use of Relief Request 4RR-10. In its verbal authorization, the NRC staff determined that the proposed alterative to defer the replacement of the existing leakoff port plug threaded connection of valve 252F007B with an ASME Code-compliant, seal-welded plug until the spring 2025 refueling outage has a minimal impact on safety. The NRC staff further determined that the proposed alternative is justified on the basis that performing the ASME Code repair in the spring 2023 refueling outage would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The staff's written evaluation below documents the technical basis for the NRC's verbal authorization.

Hardship Justification

In its evaluation, the NRC staff assessed whether the licensee provided adequate descriptions and technical information to support the basis for a hardship or unusual difficulty if it were required to perform the ASME Code repair in the spring 2023 refueling outage. The licensee's bases for the hardship are as follows:

- It is not possible to replace the existing port plug with an ASME Code-compliant plug at valve 252F007B because of the proximity of the valve to the reactor vessel and size of the line. If leakage occurs through the port or valve packing during replacement of the plug without isolating valve 252F007B, then the only way to prevent water leaking through the valve would be to place a disc on the backseat of the valve, which may not eliminate all leak by.
- For safe installation of an ASME Code-compliant plug on valve 252F007B, the licensee must offload entire core and drain the reactor vessel water below the core spray injection lines. These activities would result in an unnecessary fuel move, an increased risk of damaging fuel, and exposing personnel to unnecessary occupational hazards.
- Deferral of replacing the existing plug of valve 252F007B with an ASME Code-compliant plug for one operating cycle allows for additional planning and helps to reduce the risk of human performance errors during full core offload and vessel drain down activities.

Based on its review of the information above, the NRC staff finds the licensee's hardship justification is acceptable because the risk to plant safety and risk of personnel exposure to occupational hazards and excessive radiation constitute hardship without a compensating increase in the level of quality and safety.

Reasonable Assurance of Structural Integrity and Safe Operation for One Cycle of Operation

In its evaluation, the NRC staff assessed whether (a) there exists reasonable assurance that the structural integrity and leak tightness of the existing port plug of valve 252F007B will be maintained, and (b) the licensee has procedures in place to identify any leakage from the plug, if it were to occur, and take appropriate corrective actions for orderly shutting down of the unit. The NRC staff verified the following:

- The licensee confirmed that since the redesign of valve 252F007B in 1988, there has not been any history of leakage at the valve's port plug threaded connection. The packing of valve 252F007B has continued to be retorqued according to appropriate preventive maintenance activities, and the most recent retorque occurred in 2021.
- The licensee has performed the appropriate inspections of valve 252F007B and confirmed no evidence of leakage at the port plug.
- The licensee continues to perform the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P required system leakage test and associated visual

examination of valve 252F007B during the unit's spring 2023 refueling outage to ensure no leakage is evident prior to restart of plant operation.

- The licensee will monitor the drywell for sources of unidentified leakage in accordance with technical specifications (TS) surveillance requirement (SR) 3.4.4.1, which requires verification that the reactor coolant system leakage meets the limits specified in limiting condition for operation (LCO) 3.4.4. The frequency of the SR, which is controlled under the surveillance frequency control program, is currently set at once per 12 hours. Should leakage occur from valve 252F007B, this leakage would register during performance of SR 3.4.4.1. If any of the leakage limits are exceeded, then the TS 3.4.4 required actions require that the leakage must be reduced to within the limits in four hours or the unit must be in Mode 3 (hot shutdown) in the next 12 hours. LCO 3.4.4 is required to be met during Modes 1, 2, and 3, and will ensure valve 252F007B remains free of unacceptable leakage during Cycle 22 of operation.
- The licensee will implement Susquehanna procedure ON-DWLEAK-201, "Drywell Leakage," which prescribes a graded approach for responding to drywell leakage up to and including a unit shutdown prior to reaching the TS limits for leakage. If a presence of drywell unidentified leakage has been detected, the licensee has multiple methods to identify the source of leak. First, the licensee would analyze a sample from the drywell sump to determine if the leakage is reactor coolant leakage. Second, the licensee would evaluate the drywell temperature elements for temperature changes. Valve 252F007B is located on the 767-foot elevation of the drywell, and there are two temperature elements located on the 799-foot elevation of the drywell. These temperature elements are in the conical section of the drywell, so any temperature changes caused by a leak from the port plug of valve 252F007B would be amplified because of the reduced air volume compared to the rest of the drywell. Third, the licensee would measure the radiation inside drywell containment by the drywell containment radiation monitors, which helps to identify any sources of leak from the reactor coolant pressure boundary.

Based on the above, the NRC staff determines that no degradations and leaks have been detected in the port plug of valve 252F007B since 1988. The licensee has appropriate measures in place to detect any unidentified leakage inside drywell containment and accurately identify the source of a leak if the port plug of valve 252F007B leaks. Furthermore, the NRC staff understands that in the event of potential failure of the port plug of valve 252F007B, the amount of loss of coolant would be within the capacity of normal plant makeup and operators' ability to support an orderly shutdown of the unit.

Therefore, the NRC staff finds that (1) there is reasonable assurance that the licensee's proposed alternative has a minimal impact on public safety, and (2) the licensee's hardship justification is acceptable.

CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the leakoff port plug of the core spray injection-to-reactor vessel valve 252F007B for one operating cycle, and that complying with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of Relief Request 4RR-10 at Susquehanna, Unit 2 for one cycle (Cycle 22) of operation, which spans the remainder of the fourth and the initial portion of the fifth 10-year ISI intervals through the unit's spring 2025 refueling outage.

All other ASME Code, Section XI requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: A. Rezai, NRR
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Date: August 1, 2023

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Plant Licensing Branch I
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SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 – AUTHORIZATION AND SAFETY EVALUATION FOR RELIEF REQUEST 4RR-10 (EPID L-2023-LLR-0014) DATED AUGUST 1, 2023

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ADAMS Accession No. ML23207A176 **NRR-028**

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