**Kevin Cimorelli**
Site Vice President

**Susquehanna Nuclear, LLC**
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3795 Fax 570.542.1504
Kevin.Cimorelli@TalenEnergy.com

TALEN ENERGY

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

10 CFR 50.55a

**SUSQUEHANNA STEAM ELECTRIC STATION**
**RELIEF REQUEST 4RR-02, REVISION 1**
**PLA-7895**

**Docket No. 50-387**
**and 50-388**

*References:*

1. NRC letter to SSES, "Relief Request No. 22 (RR-22) from American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Susquehanna Steam Electric Station Units 1 and 2 (TAC Nos. MB0484 and MB0485)," February 28, 2001 (ADAMS Accession No. ML010330383)
2. NRC letter to SSES, "Susquehanna Steam Electric Station, Units 1 and 2 Relief Requests for the Fourth 10-Year Inservice Inspection Interval (TAC Nos. MF2705 through MF2714)," dated June 9, 2014 (ADAMS Accession No. ML14141A073)

Pursuant to 10 CFR 50.55a(z)(1), Susquehanna Nuclear, LLC (Susquehanna), hereby requests NRC approval of Relief Request 4RR-02, Revision 1, which proposes an alternative to the requirements of the American Society of Mechanical Engineers Code for inspection of Reactor Pressure Vessel circumferential welds.

In Reference 1, Susquehanna received approval for this alternative under the Second Inspection Interval Inservice Inspection (ISI) Program Plan as 2RR-22 for the remaining duration of the initial 40-year license. Subsequently in Reference 2, Susquehanna received approval for Alternative 4RR-02, Revision 0 as part of the Fourth Inspection Interval ISI Program Plan which authorized the extension of duration granted in 2RR-22 from the end of the initial 40-year license period through the end of the Fourth Inspection Interval. Relief Request 4RR-02, Revision 1, updates the analyses provided in the previous requests to extend the duration through the period of extended operation. Susquehanna is currently in the Fourth ISI Interval.

Susquehanna requests NRC approval by December 31, 2021.

This letter contains no new or revised regulatory commitments.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager- Nuclear Regulatory Affairs, at (570) 542-1818.

K. Cimorelli

Enclosure: Relief Request 4RR-02, Revision 1

Copy: NRC Region I
Mr. C. Highley, NRC Sr. Resident Inspector
Ms. S. Goetz, NRC Project Manager
Mr. M. Shields, PA DEP/BRP

# Enclosure to PLA-7895

# Relief Request 4RR-02, Revision 1

**Acronyms**

| Acronym | Description |
|---|---|
| ASME | American Society of Mechanical Engineers |
| BWR | Boiling Water Reactor |
| BWRVIP | Boiling Water Reactor Vessel and Internals Program |
| CFR | Code of Federal Regulations |
| CRD | Control Rod Drive |
| ECCS | Emergency Core Cooling System |
| EFPY | Effective Full Power Years |
| HPCI | High Pressure Coolant Injection |
| ISI | Inservice Inspection |
| LER | Licensee Event Report |
| LRA | License Renewal Application |
| NRC | Nuclear Regulatory Commission |
| PEO | Period of Extended Operation |
| RCIC | Reactor Core Isolation Cooling |
| RG | Regulatory Guide |
| RPV | Reactor Pressure Vessel |
| RCS | Reactor Coolant System |
| $RT_{NDT}$ | Reference Temperature (Nil-Ductility Transition) |
| RWCU | Reactor Water Cleanup |
| SE | Safety Evaluation |
| SSES | Susquehanna Steam Electric Station |
| TLAA | Time Limited Aging Analyses |
| UFSAR | Updated Final Safety Analysis Report |

*** NOTE ***

This request for alternative was previously approved for the remaining duration of the initial 40-year license on February 28, 2001, under the Second Inspection Interval ISI Program Plan as 2RR-22 (Reference 1).

Subsequently, this request for alternative was approved on June 9, 2014, under the Fourth Inspection Interval ISI Program Plan as 4RR-02 Revision 0, which extended the relief from the end of the initial 40-year license period through the end of the Fourth Inspection Interval (Reference 2).

This request for alternative presented under 4RR-02 Revision 1 updates the analyses provided in the previous requests to extend the duration through the PEO.

## ASME CODE COMPONENTS AFFECTED

Code Class: 1; Reference: ASME Section XI, Table IWB-2500-1; Examination Category B-A: Item Number, B1.11; Welds on Unit 1 and 2: Weld IDs AA, AB, AC, AD, and AE.

## CODE EDITION AND ADDENDA / APPLICABLE CODE REQUIREMENT

ASME Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, requires a volumetric examination of the circumferential shell welds each interval.

## REASON FOR REQUEST / RELIEF REQUESTED

Susquehanna Nuclear, LLC (Susquehanna), requests an alternative in accordance with 10 CFR 50.55a(z)(1) on the basis that this alternative provides an acceptable level of quality and safety. This request for alternative would provide relief from circumferential weld examinations required by the ASME Section XI Code for the PEO.

As documented in NUREG-1931 (Reference 3), the NRC's review of the Susquehanna LRA (Reference 4) concluded that SSES had demonstrated, in accordance with 10 CFR 54.21(c)(1)(ii), that the analysis for RPV circumferential weld examination relief had been projected to the end of the PEO. The NRC also concluded that the UFSAR supplement contained an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d) and therefore is acceptable.

## PROPOSED ALTERNATIVE AND BASIS FOR RELIEF

### Proposed Alternative

Susquehanna requests the use of BWRVIP-05 (Reference 5) with supporting information described herein as the bases for excluding the RPV shell circumferential welds from the examinations required by ASME Section XI, Examination Category B-A, Item No. B1.11 for the PEO ending at midnight on July 17, 2042 for Unit 1, and at midnight on March 23, 2044 for Unit 2.

The axial weld seams (Examination Category B-A, Item No. B1.12) and their intersection with the associated circumferential weld seams will be examined in accordance with ASME Section XI except where specific relief is granted when essentially complete (>90%) coverage cannot be obtained.

### Basis for Relief

The technical basis supporting the requested alternative is provided by the July 28, 1998 BWRVIP-05 SE (Reference 6). In the BWRVIP-05 SE, the NRC concluded that because the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency of any BWR plant, continued inspection would result in a negligible decrease in an already acceptably low RPV failure probability. Therefore, elimination of the ISI requirements for RPV circumferential welds is justified.

The BWRVIP-05 SE indicated that BWR applicants may request relief from ASME Code Section XI requirements for volumetric examination of circumferential RPV welds by demonstrating that (1) at the expiration of their license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds; and (2) the applicants must implement operator training and operating procedures that limit the frequency of cold overpressure events to the amount specified. The BWRVIP-05 SE also indicated that the requirements for inspection of RPV circumferential welds during an additional 20-year license renewal period would need plant-specific reassessment considering the chemistry of its limiting weld and the neutron fluence at the end of the license renewal term as part of any BWR LRA. The applicant also must request relief from the ASME Code Section XI requirements for volumetric examination of circumferential welds for the PEO in accordance with 10 CFR 50.55a(z). Susquehanna initially applied for relief from circumferential vessel shell weld volumetric examinations in 2RR-22 (Reference 7). Approval of the relief request for the remaining duration of the original 40-year license was granted by the NRC in February 2001 (Reference 1).

The Susquehanna LRA Table 4.2-8 shows that the initial $RT_{NDT}$ and $\Delta RT_{NDT}$ for the limiting circumferential weld for Unit 2 at 54 EFPY (with heat No. 624263/E204A27A) are -20 °F and 30.9 °F, resulting in a mean $RT_{NDT}$ (without the margin term) of 10.9 °F. The corresponding Unit 1 mean $RT_{NDT}$ is -0.6 °F (Reference 8). This value was updated from the Unit 1 $RT_{NDT}$ provided in the Susquehanna LRA Table 4.2-7 due to results of the last Unit 1 surveillance specimen testing results. It is also noted that the copper and nickel content, chemistry factor, and initial $RT_{NDT}$ for the limiting weld for both units as presented in 2RR-22 are consistent with those in the BWRVIP-05 SE. As such, NUREG-1931 Section 4.2.5.2 concluded that since the 54 EFPY mean $RT_{NDT}$ value for either Unit 1 or Unit 2 is less than the 64 EFPY value from the BWRVIP-05 SE, the RPV conditional failure probability for either unit at 54 EFPY is bounded by the generic analysis in the BWRVIP-05 SE. Reference Table 1 below for a synopsis of limiting weld data.

**Table 1 - SSES Circumferential Weld Evaluation for 54 EFPY**

| Parameter Description | SSES Units 1 and 2 Comparative Parameters at 54 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 624263/E204A27A* | USNRC Limiting Plant Specific Analyses Parameters at 64 EFPY (BWRVIP-05 SE Table 2.6-5) |
|---|---|---|
| Cu, wt% | 0.06 | 0.10 |
| Ni, wt% | 0.89 | 0.99 |
| CF | 82.0 | 109.5 |
| EOL ID Fluence, x $10^{19}$ n/cm$^2$ | 0.0941 | 1.02 |
| $\Delta RT_{NDT}$, °F | 30.9 | 135.6 |
| $RT_{NDT(U)}$, °F | -20 | -65 |
| Mean $RT_{NDT}$, °F | 10.9 | 70.6 |

* Unit 2 data. Unit 1 data is enveloped by this data.

For the second condition, approval of the initial relief request concluded that the SSES implementation of operator training and establishment of procedures, limiting the frequency of cold overpressure events to the frequency specified in the BWRVIP-05 SE for the remaining initial 40-year licensed period of operation was acceptable. SSES will continue to provide procedures, processes, and operator training to limit the frequency of cold overpressure events throughout the PEO.

The System Leakage Test has sufficient procedural guidance to prevent a cold over-pressurization event. The System Leakage Test is performed at the conclusion of each refueling outage. Briefing for these tests generally detail the anticipated testing evolution

with special emphasis on conservative decision making, plant safety awareness, the process in which the test would be aborted if plant systems responded in an adverse manner, and lessons learned from similar in-house or industry operating experiences. Specific attention is devoted to avoidance of rapid over-pressurization by an inadvertent SCRAM at test pressure (Reference 9). Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification 3.4.10, "RCS Pressure and Temperature (P/T) Limits," pressure-temperature curve. The procedure for this test prescribes the designation of a test director (on a shiftly basis) for the duration of the test who is a single point of accountability, responsible for the coordination of testing from initiation to closure, and for maintaining shift management and line management cognizant of the status of the test. Additionally, the Shift Supervisor provides an oversight function during the test.

Additionally, to ensure a controlled, deliberate pressure increase, the rate of pressure increase is administratively limited throughout the performance of the test. If the pressurization rate exceeds this limit, direction is provided to remove the CRD pumps, which are used for pressurization, from service.

With regard to inadvertent system injection resulting in a low-temperature overpressure condition, the high pressure make-up systems (HPCI and RCIC systems, as well as the normal feedwater supply (via the Reactor Feedwater Pumps)) are all steam driven. During reactor cold shutdown conditions, no reactor steam is available for the operation of these systems. Therefore, it is not possible for these systems to contribute to an overpressure event while the unit is in cold shutdown. Although auxiliary steam is used to test the associated turbines while the plant is shutdown, the pump is uncoupled from the turbine during the actual test which would prevent an overpressure condition.

Procedural control is also in place to respond to an unexpected or unexplained rise in reactor water level which could result from a spurious actuation of an injection system. Actions specified in this procedure include preventing condensate pump injection, securing ECCS system injection, tripping CRD pumps, terminating all other injection sources, and lowering RPV level via the RWCU system.

In addition to procedural barriers, Licensed Operator Training is in place which further reduces the possibility of the occurrence of over-pressurization events. During Initial Licensed Operator Training the following topics are covered: brittle fracture and vessel thermal stress; Technical Specification training, including Section 3.4.10; and Simulator Training of plant heat up and cooldown including performance of surveillance tests which ensure pressure-temperature curve compliance. In addition, operator training has been provided on the expectations for procedural compliance as provided in the operations standards manual.

During plant outages, the work control processes assure that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. Work activities are reviewed by Station Management and Operations Management to ensure safe operation and that plant mode can support the scheduled work.

During outages, work is coordinated through the Outage Control Center. In the Control Room, the Shift Supervisor is procedurally required to maintain cognizance of any activity that could potentially affect reactor level or decay heat removal during refueling outages. The Control Room Operators are required to provide positive control of reactor water level within the specified bands, and promptly report when operating outside the specified band, including restoration of actions being taken.

In addition to the above, ongoing review of industry operating plant experiences is conducted to ensure that the SSES procedures consider the impact of actual events, including over-pressurization events. Appropriate adjustments to the procedures and associated training are then implemented, to preclude similar situations from occurring at SSES.

Summary

The BWRVIP-05 provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds. The BWRVIP-05 concludes that the probability of failure of the BWR reactor pressure vessel circumferential shell welds is orders of magnitude lower than that of that axial shell welds. Based on an assessment of the materials in the circumferential weld in the beltline of the SSES Unit 2 RPV, the conditional probability of RPV failure should be less than or equal to that estimated in the BWRVIP-05 SE. Based on operator training and established procedures that have been implemented, the probability of cold overpressure transients will limit the frequency of cold overpressure events to the amounts specified in the BWRVIP-05 SE.

## ALTERNATE EXAMINATIONS

Susquehanna proposes to perform inspections of essentially 100 percent of the longitudinal seam welds in the RPV shell and essentially zero percent of the RPV circumferential seam welds, which will result in partial examination (i.e., approximately two to three percent) of the circumferential welds at their points of intersection with the longitudinal welds. These inspections are being proposed as an alternative to the ISI requirements for circumferential welds in ASME Section XI.

## DURATION FOR THE PROPOSED ALTERNATIVE

The proposed alternative will be applied starting with approval of this request and terminating at the end of the PEO at midnight on July 17, 2042 for Unit 1, and at midnight on March 23, 2044 for Unit 2.

## PRECEDENTS

The NRC has authorized similar requests to adopt an alternative to the ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item. No. 81 .11 criteria for permanent relief from the volumetric examination of RPV circumferential shell welds for the PEO. A similar relief request has been granted to the following plant: NRC letter, "Duane Arnold Energy Center - Request for Relief No. RR-01, Regarding Extension of Permanent Relief from Ultrasonic Examination of Reactor Pressure Vessel Circumferential Shell Welds for the Renewed Operating License Term (CAC NO. MF9380, EPID L-2017-LLR-0108), dated January 19, 2018 (ADAMS Accession No. ML17353A682).

**REFERENCES**

1. NRC letter to SSES, "Relief Request No. 22 (RR-22) from American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Susquehanna Steam Electric Station Units 1 and 2 (TAC Nos. MB0484 and MB0485)," February 28, 2001 (ADAMS Accession No. ML010330383)

2. NRC letter to SSES, "Susquehanna Steam Electric Station, Units 1 and 2 Relief Requests for the Fourth 10-Year Inservice Inspection Interval (TAC Nos. MF2705 through MF2714)," dated June 9, 2014 (ADAMS Accession No. ML14141A073)

3. NUREG-1931, "Safety Evaluation Report Related to the License Renewal of Susquehanna Steam Electric Station, Units 1 and 2," November 2009 (ADAMS Accession No. ML093170786)

4. SSES Letter to NRC, "Susquehanna Steam Electric Station Application for Renewed Operating Licenses Numbers NPF-14 and NPF-22 (PLA-6110)," dated September 13, 2006 (ADAMS Accession No. ML062630217)

5. EPRI TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995 (ADAMS Accession No. ML032200246)

6. NRC Letter to BWRVIP, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," dated July 28, 1998 (ADAMS Accession No. 9808040037 and 9808040041)

7. SSES to NRC, "Susquehanna Steam Electric Station Proposed Relief Request No. RR-22 Request for Alternative to 10CFR50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds for PPL Susquehanna LLC Units 1 and 2 (PLA-5251)," dated November 7, 2000 (ADAMS Accession No. ML003769393)

8. Calculation EC-062-1149 Revision 1, "SSES Units 1 and 2 RPV Beltline ART and USE Calculation for 40 and 54 EFPY," May 11, 2020

9. Clinton Power Station LER 89-016-00, "Inadequate Test Procedures Result in Group 1 Containment Isolation During Restoration from Control Room Scram Time Testing with the Reactor Pressure Vessel Solid," dated April 18, 1989 (ADAMS Accession No. 8904270425)