**Jon A. Franke**
Site Vice President

**PPL Susquehanna, LLC**
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.2904 Fax 570.542.1504
jfranke@pplweb.com

AUG 3 0 2013

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

**SUSQUEHANNA STEAM ELECTRIC STATION**
**PROPOSED RELIEF REQUESTS FOR THE FOURTH TEN-YEAR INSERVICE INSPECTION INTERVAL FOR SUSQUEHANNA UNITS 1 AND 2**
**PLA-7052**

**Docket Nos. 50-387 and 50-388**

Pursuant to 10CFR 50.55a(a)(3)(i) and 10CFR50.55a(a)(3)(ii), PPL Susquehanna, LLC hereby requests NRC authorization of the enclosed relief requests associated with the Fourth Ten-Year Inservice Inspection (ISI) Interval for the Susquehanna Steam Electric Station (SSES) Units 1 and 2. The Fourth Inspection Interval for the SSES ISI program will commence on June 1, 2014 and will comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2007 Edition through the 2008 Addenda.

The following Relief Requests are presented for consideration and review:

**4RR-02** - Requests an alternative in accordance with 10CFR 50.55a(a)(3)(i) to the requirements of the ASME Code, Subsection IWB for the requirement to inspect 100 percent of the required inspection volume. NOTE: This Relief Request is being provided as an administrative placeholder for the fourth 10-year inspection interval. This Relief was submitted in the Second Inspection Interval as 2RR-22. The approval of this Relief authorized under the SER dated February 28, 2001 is for the remaining initial license and thus included the Fourth Inspection Interval.

**4RR-05** - Requests an alternative in accordance with 10CFR 50.55a(a)(3)(i) to the requirements of the 1998 Edition of the ASME Code, Subsection IWB for the pressure testing of mechanical joints. In lieu of the requirements of the ASME Code, Subsection IWB, a proposed alternative is requested which provides an acceptable level of quality and safety.

**4RR-06** - Requests an alternative in accordance with 10CFR 50.55a(a)(3)(i) from the requirements of the ASME Code, Subsection IWF for the visual inspection of snubber attachments. In lieu of the requirements of the ASME Code, Subsection IWF, a proposed alternative is requested which provides an acceptable level of quality and safety.

**4RR-07** - Requests an alternative in accordance with 10CFR 50.55a(a)(3)(i) from the requirements of the ASME Code, Subsection IWC for the for the VT-2 visual examinations of the reactor vessel head required to be performed during a system leakage test. In lieu of the requirements of the ASME Code, Subsection IWC, a proposed alternative is requested which provides an acceptable level of quality and safety.

**4RR-08** - Requests an alternative in accordance with 10CFR 50.55a(a)(3)(i) from the requirements of the ASME Code, Subsection IWC for the VT-2 visual examinations of the CRD accumulators required to be performed during a system leakage test. In lieu of the requirements of the ASME Code, Subsection IWC, a proposed alternative is requested which provides an acceptable level of quality and safety.

PPL Susquehanna, LLC requests that the NRC authorize the attached proposed alternatives by April 25, 2014 to support implementation of the fourth ten-year inspection interval. The attached requests are proposed for the duration of the fourth ten-year inspection interval.

If you have any questions, please contact Mr. John L. Tripoli, Manager, Nuclear Regulatory Affairs, at (570) 542-3100.

This letter contains no new regulatory commitments.

Sincerely,

J. A. Franke

Attachments:

Attachment 1 - Relief Request 4RR-02
Attachment 2 - Relief Request 4RR-05
Attachment 3 - Relief Request 4RR-06
Attachment 4 - Relief Request 4RR-07
Attachment 5 - Relief Request 4RR-08

Copy: NRC Document Control Desk
Mr. J. E. Greives, NRC Sr. Resident Inspector
Mr. J. A. Whited, NRC Project Manager
Mr. L. J. Winker, PA DEP/BRP

# Attachment 1 to PLA-7052

# Relief Request 4RR-02

*** NOTE ***

SSES Fourth Inspection Interval Request for Alternative 4RR-02 is simply an administrative placeholder. This request for alternative was previously submitted and approved under the Second Inspection Interval ISI Program Plan as 2RR-22. The approval authorized under SER dated February 28, 2001 is for the remaining initial license and thus includes the Fourth Inspection Interval.

Formatting for Request for Alternative 2RR-22 varied from the standard format due to the fact that it also requested an alternative from the Augmented Vessel examination contained in 10 CFR 50.55a(g)(6)(ii)(A)(2).

The Request for Alternative is carried here and renumbered as 4RR-02 purely for administrative purposes. All ASME Code references were made in accordance with the 1989 Edition of ASME Section XI. No changes to the actual approved alternative have been made and no further or revised authorization is required.

## SYSTEM/COMPONENT(S) FOR WHICH RELIEF IS REQUESTED

Examination Category B-A, Item Number, B1.11 Welds on Unit 1 and 2: Weld IDs AA, AB, AC, AD, and AE.

## CODE REQUIREMENT

10 CFR 50.55a(g)(6)(ii)(A)(2) requires volumetric of RPF shell welds to be performed completely, once, as an augmented examination requirement. These examinations are required to be performed using the 1989 Edition of the ASME Code Section XI. These examinations are required during the inspection interval when the regulation was approved or the first period of the next inspection interval. For purposes of the augmented examinations the regulation defined "essentially 100 percent" as more than 90 percent of the examination volume of each weld.

## RELIEF REQUESTED

PPL requests approval of an alternative RPV examination for SSES Units 1 and 2. Approval of this alternative examination is requested in accordance with 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(g)(6)(ii)(A)(5) for permanently excluding volumetric examination of circumferential RPV welds. PPL also requests approval to implement the alternative RPV examination in lieu of the ISI requirements for circumferential welds in the ASME Code, Section XI 1989 Edition Table IWB-2500-1, Examination Category B-A, Item Number B1.11 volumetric examination of RPV circumferential welds. The Code of record for the second inspection interval is the ASME Code, Section XI, 1989 Edition.

## BASIS FOR RELIEF

In Generic Letter 98-05, the NRC stated that the estimated failure frequency of the BWR RPV circumferential welds is well below the acceptable core damage frequency (CDF) and large early release frequency (LERF) criteria discussed in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk Informed Decisions On Plan-Specific Changes to the Licensing Basis." Furthermore, the NRC indicated that the estimated frequency of RPV circumferential weld failure bounds the corresponding CDF and LERF that may result from a reactor pressure vessel weld failure. On this basis, the NRC concluded the proposal in the BWRVIP-05 report, as modified by two criteria, was acceptable and that BWR licensees may request permanent relief from the ISI requirements of 10CFR50.55a(g) for the volumetric examination of circumferential reactor welds by demonstrating the two criteria discussed below. The generic letter states that licensees still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Generic Letter 98-05 Criterion 1
At the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998 safety evaluation (of GL 98-05 Permitted Action).

PPL Response

SSES Units 1 and 2 are defined as ASTM E-185-73, Case "A" plants, since the vessels have a predicted shift in the reference nil-ductility temperature ($\Delta RT_{NDT}$) of less than 100°F and will be exposed to a neutron fluence of less than $5x10^{18}$ n/cm$^2$ over the design lifetime of the plant. The expected low RPV 1/4T 32 Effective Full Power Years (EFPY) beltline fluence ($<<5x10^{18}$ n/cm$^2$) results in a low predicted shift in the reference nil-ductility temperature RTNDT (<25°F at 32 EFPY).

The following table illustrates that the SSES Units 1 and 2 reactor pressure vessels have additional conservatism in comparison to Table 2.5-4 for the Limiting Plant-Specific Analyses (32 EFPY) of the NRC's evaluation of BWRVIP-05. The chemistry factor, $\Delta RT_{NDT}$, $RT_{NDT(U)}$ and Mean $RT_{NDT}$ are determined in accordance with the guidelines of Regulatory Guide 1.99, Rev. 2 and ASME Code Section III, NB2300, as applicable.

**BASIS FOR RELIEF** (Continued)

| Parameter Description | SSES Units 1 and 2 Comparative Parameters at 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 62463/E 204A27A* | NRC Limiting Plant Specific Analyses Parameters at 32 EFPY SER Table 2.6-4 |
|---|---|---|
| Cu, wt% | 0.06 | 0.10 |
| Ni, wt% | 0.89 | 0.99 |
| CF | 82 | 109.5 |
| EOL ID Fluence, $x10^{19}$ $n/cm^2$ | 0.078 | 0.51 |
| $\Delta RT_{NDT}$, °F | 24.9 | 109.5 |
| $RT_{NDT(U)}$ | -20 | -65 |
| Mean $RT_{NDT}$, °F | 4.9 | 44.5 |

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

The chemistry factors for the SSES Units 1 and 2 limiting circumferential welds are lower than the NRC's Limiting Plant-Specific Analyses (32 EFPY) and the End of Life (EOL) fluence is significantly lower than the NRC's limit such that the resulting shift in reference temperature, $\Delta RT_{NDT}$, is bounded by the NRC evaluation of BWRVIP-05 technical bases. Considering the expected shift in $RT_{NDT}(\Delta RT_{NDT})$ is small and the excellent SSES Units 1 and 2 plate and weld chemistry, embrittlement due to fluence effects have a negligible affect on the SSES Units 1 and 2 reactor pressure vessel weld failure probability, P (F/E), in the NRC's Limiting Plant-Specific Analyses (32EFPY).

Generic Letter 98-05 Criterion 2

Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 28, 1998 safety evaluation.

PPL Response

PPL has in place procedures which monitor and control reactor temperature and water inventory during all aspects of cold shutdown which would minimize the likelihood of a Low Temperature

**BASIS FOR RELIEF** (Continued)

Over-Pressurization (LTOP) event from occurring. Additionally, these procedures are reinforced through operator training.

The System Leakage Test and the System Hydrostatic Test (as modified by ASME Code Case N-498-1), which have been used at SSES, have sufficient procedural guidance to prevent a cold overpressurization event. The System Leakage Test is performed at the conclusion of each refueling outage, while the System Hydrostatic Test is performed once each Ten-Year Inspection Interval. Briefings 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 overpressurization by an inadvertent SCRAM at test pressure (in the manner of Clinton Power Station LER 89-016). Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification 3.4.10 pressure-temperature curve. The procedures for these tests prescribe the designation of a test director (on a shift 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 Control Rod Drive (CRD) pumps, which are used for pressurization, from service.

With regard to inadvertent system injection resulting in an LTOP condition, the high pressure make-up systems (High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, as well as the normal feedwater supply (via the Reactor Feedwater Pumps) at SSES 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 over-pressure 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 LTOP condition.

**BASIS FOR RELIEF** (Continued)

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 LTOP 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 "RCS Pressure and Temperature (P/T) Limits," and Simulator Training of plant heatup 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 schedule work.

During outages, work is coordinated through the Outage Control Center and the Ops Work Control Center which provides an additional level of Operations oversight. In the Control Room, the Shift Supervisor is required, by procedure, 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 the PPL procedures consider the impact of actual events, including LTOP events. Appropriate adjustments to the procedures and associated training are then implemented, to preclude similar situations from occurring at SSES.

<u>Summary</u>

The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds. The BWRVIP-05 report 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 reactor pressure vessel, the conditional probability of reactor pressure vessel failure should be less than or equal to that estimated in the NRC's analysis. Based on operator training and established procedures that have been implemented, the probability of cold over-pressure transients will limit the frequency of cold over-pressure events to the amounts specified in the NRC's June 30, 1998 safety evaluation.

## REFERENCES

1. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

2. EPRI TR 105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), September 1995.

3. NRC Letter from Gus C. Lainas, Acting Director, Division of Engineering, Office of Nuclear Reactor Regulation, to Carl Terry, BWRVIP chairman, Niagara Mohawk Company, July 28, 1998.

## ALTERNATE EXAMINATIONS

PPL 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 augmented examinations specified in 10 CFR 50.55a(g)(6)(ii)(A)(2) for circumferential welds, as well as an alternative to the ISI requirements for circumferential welds in the ASME Code, Section XI 1989 Edition.

## IMPLEMENTATION SCHEDULE

This relief will remain in effect for the duration of the Second 10 year interval of the ISI Program for SSES Units 1 and 2 (June 1, 2004).

Note: Relief Request 4RR-02 is provided for information purposes. Permanent relief was requested by SSES from the examination requirements of 10CFR50.55a for RPV circumferential shell welds since the proposed alternative provides an acceptable level of quality and safety. Permanent relief was authorized by the NRC in a SER dated February 28, 2001.

# Attachment 2 to PLA-7052

# Relief Request 4RR-05

## COMPONENT IDENTIFICATION

| | |
|---|---|
| Code Class: | 1 |
| Reference: | IWB-5221(a) |
| Examination Category: | Not Applicable |
| Item Number: | Not Applicable |
| Description: | Alternative to ASME Section XI, IWB-5221(a), use of Code Case N-795 |
| Component Number: | Not Applicable |

## APPLICABLE CODE EDITION AND ADDENDA

The Susquehanna Steam Electric Station's (SSES) will start the 4th 10-Year Inservice Inspection (ISI) Program Interval on June 1, 2014 and is required to follow the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (ASME Section XI), 2007 Edition through the 2008 Addenda.

## CODE REQUIREMENT

**10 CFR 50.55a(b)(2)(xxvi) requires the use of the 1998 Edition, IWA-4540(c) for pressure testing of Class 1, 2, & 3 mechanical joints**

The 1998 Edition of ASME Section XI, IWA-4540(c) states: "Mechanical joints made in installation of pressure retaining items shall be pressure tested in accordance with IWA-5211(a). Mechanical joints for component connections, piping, tubing (except heat exchanger tubing), valves, and fittings, NPS-1 and smaller, are exempt from the pressure test." Susquehanna Steam Electric Station (SSES) understands that this means a pressure test is required for a mechanical joint when a new valve or flange greater than NPS-1 is installed as part of the repair/replacement activity, and does not include those items covered by IWA-4132 "Items Rotated From Stock."

Note that the 1998 Edition, IWA-5211(a) states "a system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature." SSES has defined this to be a minimum of 1035 psig for components within the Reactor Coolant Pressure Boundary (RCPB).

The applicability for Code Case N-795 begins with the 1998 Edition with the 1999 Addenda and includes applicability to the 2007 Edition with the 2008 Addenda; although the 1998 Edition specified in 10 CFR 50.55a(b)(2)(xxvi) is not included in the published applicability, SSES believes that the comparison of IWB-5211(a) from the 1998 Edition and IWB-5221(a) of the 2007 Edition with the 2008 Addenda is compatible when the pressure has been defined specifically for the SSES as described above. Therefore, SSES concludes that Code Case N-795 may be used for the 1998 Edition specified by the NRC condition found in 10 CFR 50.55a.

**<u>Welded or Brazed Joints</u>**

**ASME Section XI, 2007 Edition with the 2008 Addenda**

The 2007 Edition with the 2008 Addenda, IWA-4540(a) states: "Unless exempted by IWA-4540(b), repair/replacement activities performed by welding or brazing on pressure-retaining boundary shall include a hydrostatic or system leakage test in accordance with IWA-5000, prior to, or as part of, returning to service. Only brazed joints and welds made in the course of a repair/replacement activity require pressurization and VT-2 visual examination during the test."

**<u>Pressure Testing Requirements</u>**

**ASME Section XI, 2007 Edition with the 2008 Addenda**

The 2007 Edition with the 2008 Addenda, IWB-5221(a) states: "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

**<u>REASON FOR REQUEST</u>**

At the Susquehanna Steam Electric Station Units 1 and 2, Class 1 pressure tests for repair/replacement activities in accordance with IWA-4540 at pressures corresponding to 100% rated reactor power when performed after Table IWA-2500-1, Category B-P testing has been completed, requires abnormal plant conditions/alignments. Testing at these abnormal plant conditions/alignments results in additional risks and delays while providing little added benefit beyond tests which could be performed at slightly reduced pressures under normal plant conditions.

Code Case N-795 is intended to provide alternative test pressure for certain Class 1 pressure tests. The code case would be used following repair/replacement activities (excluding those on the reactor vessel) which occur subsequent to the periodic Class 1 pressure test required by Table IWB-2500-1, Category B-P and prior to the next refueling outage on those components that cannot be isolated. Components which can be isolated will be pressure tested at a pressure in accordance with IWB-5221(a).

Performance of the Category B-P pressure test each refueling outage, places SSES in a position of significantly reduced margin, approaching the fracture toughness limits defined in the Technical Specification Pressure Temperature (P-T) Curves. To violate these curves would place the vessel in a low temperature over pressure (LTOP) condition. With strict operational control procedures, specific component alignment and operations staff training regarding LTOP this may be considered acceptable to be at this reduced margin condition for the purpose of verifying the leakage status/integrity of the primary system in order to meet the ASME Section XI, Category B-P requirements prior to startup from a refueling outage, however to perform this evolution more frequently would increase the overall risks to the plant.

## PROPOSED ALTERNATIVE AND BASIS FOR USE

### Proposed Alternative:

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

SSES proposes to perform the system leakage testing and associated VT-2 examination following repair/replacement activities on those components that cannot be isolated in accordance with Code Case N-795, however using a longer hold time than specified in the code case. The system leakage test will be performed during the normal operational start-up sequence at a minimum of 932 psig (90% of the pressure required by IWB-5221(a)) following a one hour hold time (for uninsulated components) and an eight hour hold time (for insulated components) in lieu of the nominal operating pressure associated with 100% reactor power of approximately 1035 psig. Note that this code case is not applicable to Class 1 pressure tests performed to satisfy the periodic requirement of Table IWB-2500-1, Category B-P and is not applicable to pressure tests required following repair/replacement activities on the reactor vessel. SSES will continue to conduct the periodic system leakage tests required by IWB-2500-1, Category B-P at the end of each refueling outage at a pressure corresponding to 100% rated reactor power.

### Basis for Use:

By the end of a normal refueling outage the core decay heat has had time to decrease and some spent fuel has been removed and some new fuel has been added. The result is a much lower decay heat load and much lower heatup rates. At the end of a normal refueling outage, the rate of temperature increase is able to be tolerated during the system leakage test. During normal performance of this system leakage test, the pressurization phase of the test is taken at a slow and very controlled pace. The pressurization phase normally takes several hours to reach test conditions.

However, following a maintenance or forced outage, there is a much larger decay heat load from the reactor core. Once SDC is removed from service, heatup starts immediately and control of the heat load is challenged. During a short term mid-cycle shutdown, the core does have a large decay heat load with projected heatup rates in the order of 0.5°F per minute. Under those conditions, the time available to pressurize up to test conditions, perform the VT-2 exam and return to SDC will be greatly reduced. The hurried time frames may create a more error-likely environment. Considering only the actions of isolating SDC from the vessel under high decay heat loads, there is some inherent risk. There would be some probability that once isolated, mechanical, control or operational problems could occur which could delay return to SDC.

The required VT-2 examinations performed following repair/replacement activities are limited to the areas affected by the work thereby allowing for a focused exam. The VT-2 exams, therefore, have a much smaller examination boundary than the periodic test.

Indication of leakage identified through the VT-2 examinations during a test at either the 100% rated reactor power level or at 90% of that value will not be significantly different between the two tests. Higher pressure under the otherwise same conditions will produce a higher flow rate but the difference is not significant. Code Case N-795 proposes increased hold times, as compared to a test performed at normal operating pressure, to allow for more leakage from the pressure boundary if a through-wall or mechanical joint leakage condition exists; Further, SSES proposes to implement longer hold times than specified by the Code Case. SSES believes these longer hold times are justified to allow for additional leakage to accumulate at the area of interest so as to be more evident during the VT-2 examination, should a through-wall or mechanical joint leakage condition exist. This alternate test pressure, when combined with longer hold times, is still adequate to provide evidence of leakage, should a leak exist.

With respect to using the alternative requirements of Code Case N-795 to welded repair/replacement activities, the ASME concluded during the development of Code Case N-416 "Alternative Pressure Test Requirements for Welded or Brazed Repair, Fabrication Welds or Brazed Joints for Replacement Parts and Piping Subassemblies, or Installation of Replacement Items by Welding or Brazing, Classes 1, 2, and 3" and Code Case N-498, "Alternative Requirements for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems", that the hydrostatic test (a test using pressure higher than a system leakage test) was not a structural integrity test, but a leakage test. The fact that the hydrostatic test does not verify structural integrity served as the basis for replacing it with a system leakage test. Both code cases are approved by the NRC in Regulatory Guide 1.147. It is the requirements of the construction code including the construction code nondestructive examinations used for the repair/replacement activity that ensure structural integrity of the pressure boundary and its welded or brazed connections. Based on research performed by ASME, the effect of testing at a pressure that corresponds with 90% of rated power verses 100% of rated power is not reduced validation of structural integrity, but a potential in leakage rate reduction. Therefore, SSES believes that the alternative requirements of Code Case N-795 on welded or brazed repair/replacement activities are acceptable.

Research described in the White Paper performed by Argonne National Laboratory, as commissioned by the NRC, indicates that the relationship of leakage and pressure is relatively linear. Therefore, leakage rates associated with pressure at 90% of normal operating pressure would be approximately 10% less than a leakage rate at 100% of normal operating pressure. However, any reduction in leakage rate is more than compensated for by the increase in hold time (600% for noninsulated and 200% for insulated). Other research cited in the White Paper supports the conclusions of Argonne National Laboratory.

While SSES does not expect that leakage will occur, any leakage will be related to the differential pressure at the point of leakage, or across the connection. A 10% reduction in the test pressure is not expected to result in the arrest of a leak that would occur at nominal operating pressure. In the unlikely event that leakage would occur subsequent to the VT-2 examination, at higher pressures associated with 100% rated reactor power, leakage would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system (CAM), and the drywell floor drain sumps. Leakage monitoring is required by Technical Specifications.

Code Case N-795 and the SSES proposed hold times allows for an adequate pressure test to be performed; ensuring the safety margin is not reduced due to VT-2 examination being performed at the slightly reduced pressure. There is no physical benefit withheld by testing at the slightly reduced pressure. The affected pressure boundary will be tested and will be otherwise fully capable of performing its intended safety function as part of the Reactor Coolant Pressure Boundary.

The use of Code Case N-795 will only be applied if the System Leakage Test required by IWB-2500-1, Category B-P has been completed for the cycle on components that cannot be isolated and will not be implemented for any repair/replacement activity performed on the reactor pressure vessel.

In summary, the proposed alternative is to perform the system leakage test and VT-2 examination in accordance with Code Case N-795 at 932 psig with a minimum hold time of one hour for uninsulated components and an eight hour hold time for insulated components during maintenance, forced outages, or following the performance of the periodic pressure test required by Table IWB-2500-1, Category B-P during refueling outages. The provisions of this alternative are not applicable to the Examination Category B-P pressure test performed during refueling outages or to pressure tests of repair/replacement activities of the reactor pressure vessel or components that can be isolated. Considering the discussion above, SSES believe that this alternative will provide an acceptable verification of the leak integrity of the locations having repair/replacement activities performed without putting the plant in a non-conservative operational condition and without unnecessary radiation exposure and safety challenges to personnel.

## DURATION OF PROPOSED ALTERNATIVE

This relief will remain in effect for the duration of the Fourth 10 year interval of the ISI Program for SSES Units 1 and 2 (June 1, 2024).

## PRECEDENTS

A similar 10 CFR 50.55a request (Reference 1) was approved for the MNGP during their Fourth 10-Year Inservice Inspection Interval as a one-time relief by NRC letter "Monticello Nuclear Generating Plant – One Time Inservice Inspection Program Plan Relief Request No. 8 for Leak Testing the "B" and "G" Main Steam Safety Relief Valves (TAC No. MB96380)," dated June 13, 2003. (Reference 2).

# Attachment 3 to PLA-7052

# Relief Request 4RR-06

## COMPONENT IDENTIFICATION

| | |
|---|---|
| Code Class: | 1, 2, and 3 |
| Reference: | IWF-2500-1 Table |
| Examination Category: | F-A |
| Item Number: | F1.10, F1.20, F1.30, and F1.40 |
| Description: | Alternative Examination of Snubber Attachments |
| Component Number: | All Class 1, 2, and 3 Snubber Attachments |

## APPLICABLE CODE EDITION AND ADDENDA

The Susquehanna Steam Electric Station's (SSES) Units 1 and 2 will start the 4th 10-Year Inservice Inspection (ISI) Program Interval on June 1, 2014 and is required to follow the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (ASME Section XI), 2007 Edition through the 2008 Addenda.

## CODE REQUIREMENT

The 2007 Edition through the 2008 Addenda of ASME Section XI contains Figure IWF-1300-1(f) which depicts the examination boundaries for snubbers. The boundaries indicate that the attachment of the snubber to the pressure boundary and building structure is required to be examined in accordance with IWF-2000.

Table IWF-2500-1 requires a VT-3 visual examination of Class 1 (F1.10), Class 2 (F1.20), Class 3 (F1.30) piping supports, and Class 1, 2, and 3, (F1.40) component supports. The percentages for each Class are also identified: Class 1 (25%), Class 2 (15%), and Class 3 (10%). The total percentage sample shall be comprised of supports from each system (such as Main Steam, Feedwater, or RHR), where the individual sample sizes is proportional to the total number of non-exempt supports of each type and function within each system.

## REASON FOR REQUEST

Snubbers were removed from ASME Section XI in the 2006 Addenda. Figure IWF-1300-1(f) was added to show the examination boundaries for snubbers which excluded the snubber including the pivot and clevis pins (see Figure 1 below).

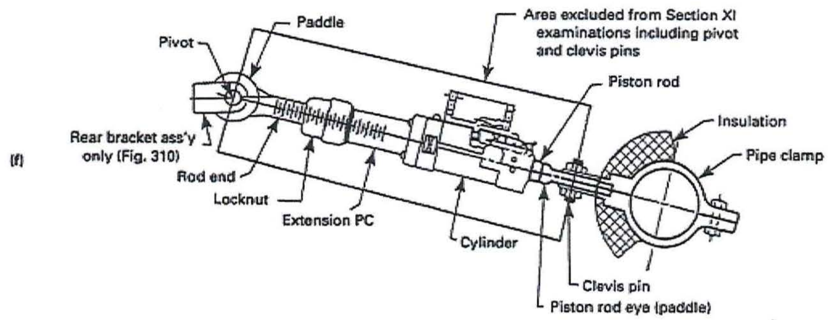



Figure 1

The attachments for the snubber to the pressure boundary (via pipe clamps etc.) and to the building structure are still included as part of the ASME Section XI examination boundary. This means that both the Snubber Program and the ISI Program requires tracking and scheduling two different examination boundaries for one component.

In order to eliminate the duplication of effort by tracking two different examination boundaries for one component, SSES requests incorporating the both examination boundaries as shown in the Figure 2 below into the Snubber Program. In addition, incorporating both examination boundaries into one program provides a better understanding of the condition of the snubber and its associated attachment to the pressure boundary or building structure. A 100% visual examination of all safety related snubbers will be performed on an examination frequency determined by the O&M Code 2004 Edition through the 2006 Addenda and Code Case OMN-13 (Note that Code Case OMN-13 has been found acceptable in RG 1.192). The examination method used for the snubber and their attachments will be the VT-3 visual examination in accordance with ASME Section XI, IWA-2213.

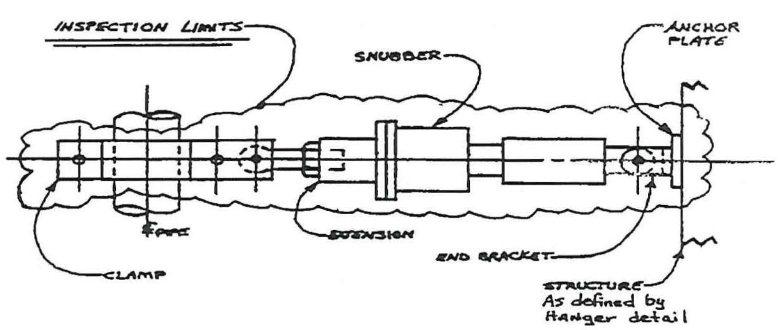



Figure 2

## PROPOSED ALTERNATIVE AND BASIS FOR USE

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety.

The O&M Code Case OMN-13 requires 100% safety related snubbers to be examined and evaluated at least once every 10 years. This exceeds the requirements of ASME Section XI, IWF-2500-1 tables which only requires 25% of Class 1, 15% of Class 2, and 10% of Class 3 required over a 10-year interval.

Performing both examination boundaries in the Snubber Program using VT-3 qualified personnel to perform the examinations provides a better understanding of the snubber and attachments. This will meet both the O&M visual examination and ASME Section XI examination requirements. This reduces the number of required examinations and personnel required to accomplish both requirements with one examination. Performing the examination on all snubber attachments in accordance with the O&M Code frequency exceeds the required percentage requirements of ASME Section XI.

## DURATION OF PROPOSED ALTERNATIVE

This relief will remain in effect for the duration of the Fourth 10 year interval of the ISI Program for SSES Units 1 and 2 (June 1, 2024).

## PRECEDENTS

None

# Attachment 4 to PLA-7052

# Relief Request 4RR-07

**RELIEF REQUEST NUMBER: 4RR-07**
**REVISION 0**


## COMPONENT IDENTIFICATION

| | |
|---|---|
| Code Class: | 2 |
| Reference: | Table IWC-2500-1 |
| Examination Category: | C-H |
| Item Number: | C7.10 |
| Description: | Request for Alternative from Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System |
| Component Number: | Flange Seal Leak Detection Line Pressure Retaining Components |

## APPLICABLE CODE EDITION AND ADDENDA

The Susquehanna Steam Electric Station's will start the 4th 10-Year Inservice Inspection (ISI) Program Interval on June 1, 2014 and is required to follow the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (ASME Section XI), 2007 Edition through the 2008 Addenda.

## CODE REQUIREMENT

Table IWC-2500-1 requires a VT-2 visual examination to be performed during a system leakage test each inspection period.

## BASIS FOR RELIEF

Pursuant to 10CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety.

The Reactor Pressure Vessel Head Flange Leak Detection Line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange (See Figure 4RR-07.1). This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap (See Figure 4RR-07.1), combined with the small size of the tap and the high test pressure requirement (1035 psig minimum), prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16 of an inch in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

A pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips. The retainer clips are contained in a recessed cavity in the top

**<u>BASIS FOR RELIEF</u>** (Continued)

head (See Figure 4RR-07.1). If a pressure test was performed from the leak-off line side with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is a thin silver plating and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that preclude this testing it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1000 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1000 psig.

System leakage testing of this line is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. It is extremely impractical to purposely fail the inner O-ring in order to perform a test.

Based on the above, SSES requests relief from the ASME Section XI requirements for system leakage testing of the Reactor Pressure Vessel Head Flange Seal Leak Detection System.

## PROPOSED ALTERNATE EXAMINATIONS

A VT-2 visual examination will be performed on the line after the refueling cavity has been filled to its normal refueling water level for at least 4 hours. The static head developed due to the water above the vessel flange during flood-up will allow for the detection of any gross indications in the line. This examination will be performed with the frequency specified by Table IWC-2500-1 for a System Leakage Test (once each inspection period).

## DURATION OF PROPOSED ALTERNATIVE

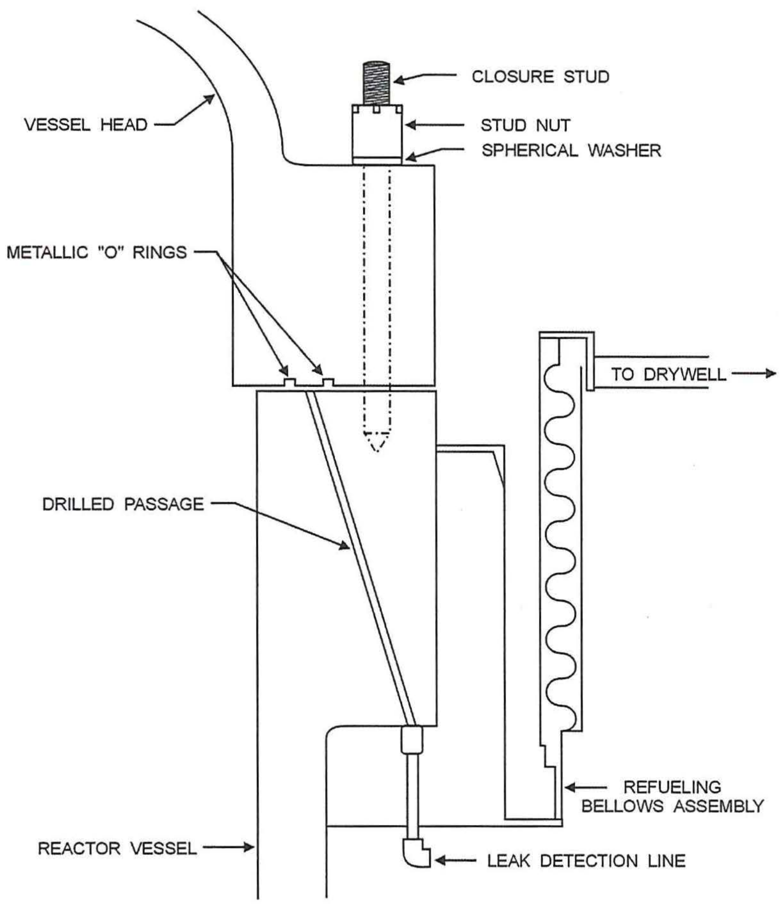
This relief will remain in effect for the duration of the Fourth 10 year interval of the ISI Program for SSES Units 1 and 2 (June 1, 2024).

## PRECEDENTS

None

**FIGURE 4RR-07.1**

# FLANGE SEAL LEAK DETECTION LINE DETAIL



# Attachment 5 to PLA-7052

# Relief Request 4RR-08

**RELIEF REQUEST NUMBER: 4RR-08**
**REVISION 0**


## COMPONENT IDENTIFICATION

| | |
|---|---|
| Code Class: | 2 |
| Reference: | Table IWC-2500-1 |
| Examination Category: | C-H |
| Item Number: | C7.10 |
| Description: | Continuous Pressure Monitoring of the Control Rod Drive (CRD) Accumulators |
| Component Number: | CRD Accumulators and Associated Piping |

## APPLICABLE CODE EDITION AND ADDENDA

The Susquehanna Steam Electric Station's will start the 4th 10-Year Inservice Inspection (ISI) Program Interval on June 1, 2014 and is required to follow the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," (ASME Section XI), 2007 Edition through the 2008 Addenda.

## CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED

Table IWC-2500-1 requires a VT-2 visual examination to be performed during a system leakage test each inspection period.

## BASIS FOR RELIEF

Pursuant to 10 CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety.

As required by the SSES Technical Specifications, the CRD Accumulator Pressure must be greater than or equal to 940 psig. Once a week, the accumulator pressure is verified for each accumulator in accordance with SSES Technical Specifications. Additionally, the accumulator pressure is continuously monitored by system instrumentation. Since the accumulators are isolated from the source of makeup nitrogen, continuous monitoring of the CRD Accumulators serves as a pressure decay type test. Should accumulator pressure fall below approximately 980 psig, an alarm is received in the control room. The pressure for the accumulator is recorded and the accumulator is recharged and checked for leaks in accordance with SSES procedures. Should a leak be detected, corrective actions are taken to repair the leak in accordance with SSES procedures.

Since monitoring the nitrogen side of the accumulators is continuous, any leakage from the accumulator would be detected by normal system instrumentation. An additional VT-2 visual examination performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a VT-2 visual examination would require applying a leak detection solution to 185 accumulators per Unit resulting in

**BASIS FOR RELIEF** (Continued)

additional radiation exposure without any added benefit in safety. This inspection would not be consistent with ALARA practices.

Relief is requested from the VT-2 visual examination requirements specified in Table IWC-2500-1 for the nitrogen side of the CRD Accumulators on the basis that continuous monitoring of the accumulator pressure and a Technical Specification required walkdown of each accumulator exceed the ASME Section XI requirement for a VT-2 visual examination.

## PROPOSED ALTERNATE EXAMINATIONS

As an alternate to the VT-2 visual examination requirements of Table IWC-2500-1, SSES will perform continuous pressure decay monitoring and a weekly Technical Specification required walkdown for the nitrogen side of the CRD accumulators including the attached piping. This alternative is similar to that contained in NRC Approved Code Case N-731 which is used for portions of Class 2 systems that are continuously pressurized by a statically-pressurized passive safety injection system.

## DURATION OF PROPOSED ALTERNATIVE

This relief will remain in effect for the duration of the Fourth 10 year interval of the ISI Program for SSES Units 1 and 2 (June 1, 2024).

## PRECEDENTS

None