Robert G. Byram
Senior Vice President and
Chief Nuclear Officer

PPL Susquehanna, LLC
Two North Ninth Street
Allentown, PA 18101-1179
Tel. 610.774.7502 Fax 610.774.5019
rgbyram@pplweb.com

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U. S. Nuclear Regulatory Commission
Attn.: Document Control Center
Mail Station OP1-17
Washington, D. C. 20555

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

Docket Nos. 50-387
and 50-388

PPL Susquehanna, LLC (PPL) requests approval of an alternative reactor pressure vessel examination for Susquehanna Steam Electric Station (SSES) Units 1 and 2. Approval of this alternative examination is requested in accordance with 10CFR50.55a(a)(3)(i) and 10CFR50.55a(g)(6)(ii)(A)(5) for permanently excluding volumetric examination of circumferential reactor pressure vessel welds. The alternative is consistent with guidance contained in Generic Letter 98-05.

PPL also requests to implement the alternative reactor pressure vessel examination in lieu of the inservice inspection requirements for circumferential welds in the ASME Boiler and Pressure Vessel Code, Section XI. The code of record for the second 10-year inservice inspection interval is the ASME Code, Section XI 1989 Edition.

On November 10, 1998, the NRC issued Generic Letter (GL) 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." GL 98-05 informed licensees of BWRs that the NRC staff had completed its review of the BWR Vessel and Internals Project (BWRVIP) BWR reactor pressure vessel shell weld inspection recommendations as contained in the BWRVIP-05 report. The GL also informed licensees that they may request permanent relief from the inservice inspection (ISI) requirements of 10CFR50.55a(g) for the volumetric examination of circumferential RPV welds provided:

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 evaluation; and

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2. Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's safety evaluation.

PPL has demonstrated in the attachment that it meets these two criteria. PPL will still perform the required inspections of "essentially 100 percent" of all axial welds.

This relief request is similar to the following requests approved by the NRC:

- GPU Nuclear, Inc. for Oyster Creek Nuclear Generating Station submitted December 30, 1999 and approved September 14, 2000.

- Tennessee Valley Authority, for Browns Ferry Nuclear Plant Unit 3, submitted June 25, 1999 and approved November 18, 1999.

PPL is currently scheduled to begin its next Unit 2 refueling outage in March 2001. PPL requests the NRC review and approve this relief request by January 15, 2001.

Should you have any questions regarding this submittal, please contact Ms. Carolyn Cino at 610-774-7614.

Sincerely,

R. G. Byram

Attachment

cc: Regional Administrator—Region 1
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Sr. Project Manager

PPL Susquehanna, LLC
Susquehanna SES Units 1 and 2
Second 10-Year Interval

RELIEF REQUEST RR-22

SYSTEM/COMPONENT(S) FOR WHICH RELIEF IS REQUESTED

Category B-A, Item No. B1.11 Welds on Units 1 and 2: Weld Ids AA, AB, AC, AD, AE

CODE REQUIREMENTS

10CFR50.55a(g)(6)(ii)(A)(2) requires volumetric examination of RPV 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 10CFR50.55a(a)(3)(i) and 10CFR 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 inservice inspection requirements for circumferential welds in the ASME code, Section XI 1989 Edition Table IWB-2500-1, Examination Category B-A, Item No. B1.11 volumetric examination of RPV circumferential welds. The code of record for the second inservice 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 Plant-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 inservice inspection 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 (ΔRT_{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 RT_{NDT} (<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 2.6-4 for the Limiting Plant-Specific Analyses (32 EFPY) of the NRC's evaluation of BWRVIP-05. The chemistry factor, ΔRT_{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.

Parameter Description	**SSES Units 1 and 2 Comparative Parameters at 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 624263/E 204A27A***	**USNRC 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
ΔRT_{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, ΔRT_{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 probabilities, which based on the above, are considered bounded by the conditional failure probability, P (F/E), in the NRC's Limiting Plant-Specific Analyses (32 EFPY).

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 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 Susquehanna SES 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.

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 scheduled 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 that 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 Susquehanna SES.

Summary

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 the axial shell welds. Based on an assessment of the materials in the circumferential weld in the beltline of the SSES Unit 2 reactor pressure vessels, 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.

RELIEF REQUEST RR-22

ALTERNATIVE 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 10CRFR50.55a(g)(6)(ii)(A)(2) for circumferential welds, as well as an alternative to the inservice inspection requirements for circumferential welds in the ASME Code, Section XI 1989 Edition.

IMPLEMENTATION SCHEDULE

PPL Susquehanna, LLC requests that this relief request be approved by January 15, 2001, in order to support the Unit 2 10^{th} Refuel Outage that is scheduled to begin in March 2001. This relief will remain in effect for the duration of the Second 10 year interval of the Inservice Inspection Program for Susquehanna SES Units 1 and 2 (June 1, 2004).