February 28, 2001

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: 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)

Dear Mr. Byram:

By letter dated November 7, 2000, PPL Susquehanna, LLC, submitted RR-22 to request relief from the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for the Susquehanna Steam Electric Station, Units 1 and 2, second 10-year inservice inspection (ISI) interval. This request for relief proposed the use of alternative inspections to perform circumferential shell weld examinations on the reactor pressure vessel welds. These examinations are required by the augmented examination requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(6)(ii)(A)(2). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and is consistent with the guidance provided in 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, and the Nuclear Regulatory Commission (NRC) staff's evaluation of the BWRVIP-05 Report issued July 28, 1998. The NRC staff has reviewed your request, and, based on the information provided, concludes that the proposed alternative will provide an acceptable level of quality and safety for the remaining term of operation under the initial existing license. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i). Our detailed evaluation and conclusions are documented in the enclosed safety evaluation.

Sincerely,

/RA/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure: Safety Evaluation

cc w/encl: See next page

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<u>DISTRIBUTION:</u>

PUBLIC	M. Gamberoni	ACRS	C. E. Carpenter
PDI-1 Reading	R. Schaaf	OGC	C. Cowgill, RGN-I
E. Adensam	M. O'Brien	J. Shea	G. Hill(4)
K. Wichman			

ADAMS ACCESSION NUMBER: ML010330383 * See previous concurrence.

OFFICE	PDI-1/PM	PDI-2/LA	EMCB/SC	OGC	PDI-1/SC
NAME	RSchaaf	MO'Brien	KWichman*	RHoefling*	MGamberoni
DATE	02/ 28 /01	02/28/01	02/12/01	02/22/01	02/28/01

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Susquehanna Steam Electric Station,
Units 1 &2

Bryan A. Snapp, Esq
Assoc. General Counsel
PPL Services Corporation
2 North Ninth Street GENTW3
Allentown, PA 18101-1179

Rocky R. Sgarro
Supervisor-Nuclear Licensing
PPL Susequehanna, LLC
2 North Ninth Street GENTW3
Allentown, PA 18101-1179

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 35, NUCSA4
Berwick, PA 18603-0035

Director-Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P.O. Box 8469
Harrisburg, PA 17105-8469

PPL Susequehanna, LLC
Nuclear Records
Attn: G. DallaPalu
2 North Ninth Street GENA62
Allentown, PA 18101-1179

Richard W. Osborne
Allegheny Electric Cooperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, PA 17108-1266

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Bryce L. Shriver
Vice President-Nuclear Site Operations
Susquehanna Steam Electric Station
PPL Susquehanna, LLC
Box 467, NUCSA4
Berwick, PA 18603-0035

Herrbert D. Woodeshick
Special Office of the President
PPL Susquehanna, LLC
Rural Route 1, Box 1797
Berwick, PA 18603-0035

George T. Jones
Vice President-Nuclear Engineering &
Support
PPL Susquehannn, LLC
2 North Ninth Street, GENA61
Allentown, PA 18101-1179

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
443 Orlando Avenue
State College, PA 16803

Board of Supervisors
Salem Township
P.O. Box 405
Berwick, PA 18603-0035

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE SECOND 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NO. 22 (RR-22)

PPL SUSQUEHANNA, LLC

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter dated November 7, 2000, PPL Susquehanna, LLC (the licensee), requested that the Nuclear Regulatory Commission (NRC) approve an alternative to performing circumferential shell weld examinations on the reactor pressure vessel (RPV) welds for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. These examinations are required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and by the augmented examination requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(6)(ii)(A)(2). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and is consistent with the guidance provided in 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, and the NRC staff's evaluation of the BWRVIP-05 Report issued July 28, 1998.

2.0 BACKGROUND

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest Edition and Addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

ENCLOSURE

3.0 EVALUATION

3.1 Relief Request No. 22, Alternatives for Examination of Reactor Pressure Vessel Shell Welds

3.1.1 Code Requirements

Section 50.55a(g)(6)(ii)(A) of 10 CFR requires that licensees perform an augmented RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code. The final Rule was published in the *Federal Register* on August 6, 1992 (57 FR 34666). By incorporating into the regulations the 1989 Edition of the ASME Code, the NRC staff required that licensees perform volumetric examinations of "essentially 100 percent" of the RPV pressure-retaining shell weld, during all inspection intervals. Section 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.1.2 BWRVIP-05 Report

By letter dated September 28, 1995, as modified and supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13 and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), submitted the proprietary report BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations." As modified, the BWRVIP report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of essentially 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RPV shell welds, except at the intersection of the axial and circumferential welds, thereby including about 2-3 percent of the circumferential welds. In addition, the report provided proposals to revise ASME Code requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420(b) of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued a safety evaluation (SE) of the BWRVIP-05 Report. This evaluation concluded that the failure frequency of RPV circumferential welds in BWRs was sufficiently low to justify elimination of inservice inspection (ISI) of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation required that examination of the RPV circumferential shell welds should be performed if axial weld examinations revealed an active, mechanistic mode of degradation.

In the BWRVIP-05 Report, the BWRVIP concluded that the conditional probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than those of the longitudinal welds. As a part of its review of the report, the NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 Report. The NRC staff's assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential welds during the (current) initial 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant, as indicated in Tables 2.6-4 and 2.6-5, respectively, of the NRC staff's July 28, 1998, SE.

The failure frequency for a reactor pressure vessel is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The NRC staff determined the conditional probability of failure for longitudinal and circumferential welds in BWR vessels fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering, and Babcock and Wilcox. The analysis identified a cold over-pressure event in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The NRC staff estimated that the probability for the occurrence of the limiting over pressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the NRC staff's evaluation identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

3.1.3 Generic Letter 98-05

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 stated that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-I, Examination Category B-A, Item 1.11, "Circumferential Shell Welds") upon demonstrating that:

(1) at the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's SE dated July 28, 1998, and

(2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's SE dated July 28, 1998.

Licensees would still need to perform the required inspections of "essentially 100 percent" of all axial welds.

3.1.4 Specific Relief Requested

The licensee identifies the following Code requirements from which relief is sought:

(1) ASME Section XI, 1989 Edition (no addenda), Table IWB-2500-1, Examination Category B-A, Item No. B1.11, volumetric examination of reactor pressure vessel circumferential welds. Permanent relief (i.e., for the remaining term of operation under the existing license) is requested.

The requested permanent relief from the Table IWB-2500-1 requirements applies to:

ISI Class 1, Code Category B-A, "Pressure Retaining Welds in Reactor Vessel," Item B1.11, "Circumferential Shell Welds" (RPV Circumferential Shell Welds: AA, AB, AC, AD, and AE).

3.1.5 Licensee's Basis for Relief

The licensee's request is based upon provisions in the NRC's SE for the BWRVIP-05 Report and the guidance outlined in GL 98-05. These documents provide the basis for the elimination of ISI of BWR RPV circumferential shell welds. As described previously, GL 98-05 provides two criteria that relief request applicants must demonstrate. One criterion is based upon the limiting conditional failure probability of the applicant's circumferential welds. The other criterion is based upon the implementation of operator training and establishment of procedures to limit the frequency of cold over-pressure events. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the SE.

3.1.6 Licensee's Evaluation of Limiting Conditional Failure Probability

The NRC SE for the BWRVIP-05 Report evaluated the conditional failure probability of circumferential welds for the limiting plant-specific case of BWR RPVs manufactured by different vendors, including CB&I, using the highest mean irradiated RT_{NDT} to determine the limiting case.

Table 1: Comparison of SSES Units 1 and 2 Bounding Circumferential Weld and the NRC Limiting Plant-Specific Analysis from Table 2.6-4 of the Final SE of the BWRVIP-05 Report

PARAMETER	SSES UNIT 2* BOUNDING CIRC. WELD AT 32 EFPY WIRE HEAT/LOT 624263/E204A207A	NRC LIMITING PLANT-SPECIFIC ANALYSIS DATA AT 32 EFPY CB&I
Cu (Wt. %)	0.06	0.10
Ni (Wt. %)	0.89	0.99
Chemistry Factor (°F)	82	109.5
EOL Fluence (10^{19} n/cm^2)	0.078	0.51
ΔRT_{NDT} (°F)	24.9	109.5
Initial RT_{NDT} (°F)	-20	-65
Mean RT_{NDT} (°F) [Initial RT_{NDT} + ΔRT_{NDT}]	4.9	44.5
* Unit 1 data is enveloped by Unit 2 data.		

The SSES Units 1 and 2 RPVs were fabricated by CB&I; therefore, the relief request compared the mean RT_{NDT} at 32 effective full-power years (EFPY) for SSES Units 1 and 2 to that for the limiting CB&I case described in Table 2.6-4 of the NRC staff's SE of the BWRVIP-05 Report. As illustrated in Table 1 of this SE, the mean RT_{NDT} for SSES Units 1 and 2 is lower than that

for the limiting CB&I case, and the licensee concluded that the conditional failure probability for the SSES Units 1 and 2 circumferential welds is bounded by the conditional failure probabilities in the NRC staff's SE through the end of the current license period.

3.1.7 Licensee's Evaluation of the Probability of Cold Overpressure Transients

During review of the BWRVIP-05 Report, the NRC staff identified non-design basis events which should have been considered in the BWRVIP-05 Report. In particular, the potential for and consequences of cold over-pressure transients should be considered. The licensee has provided its assessment of the procedures and operator training in place to monitor and control reactor temperature and water inventory during all aspects of cold shutdown to minimize the likelihood of a cold over-pressure event from occurring.

The system leakage and hydrostatic tests at SSES have sufficient procedural guidance to prevent a cold over-pressurization event. Licensee briefings for these tests include special emphasis on the criteria for aborting the test if plant systems respond in an adverse manner and lessons learned from in-house and industry operating experience. Vessel temperature and pressure are monitored throughout these tests to ensure compliance with the Technical Specification 3.4.10 pressure-temperature curve. The control rod drive system is used to control pressurization during testing. The rate of pressure increase is administratively limited throughout the performance of the test, minimizing the likelihood of exceeding the pressure-temperature limits during the test.

With regard to inadvertent system injection resulting in a cold over-pressurization event, the high pressure make-up systems (high pressure coolant injection and reactor core isolation cooling 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 turbine test, eliminating the potential for inadvertent system injection. Procedural controls are also in place to respond to an unanticipated rise in reactor water level that could result from a spurious actuation of an injection system.

The NRC staff notes that inadvertent actuation of the standby liquid control (SLC) system could also pose a cold over-pressurization risk. According to the SSES Final Safety Analysis Report, Section 9.3.5, SLC injection requires operator action to manually start the system from the control room using a key lock controller. In the event of manual initiation during shutdown, the maximum SLC injection rate of approximately 82 gpm would allow operators sufficient time to control reactor pressure in accordance with the procedural controls noted above.

In addition to procedural barriers, licensed operator training is in place which further reduces the possibility of the occurrence of cold over-pressurization events. Training includes brittle fracture and vessel thermal stress, TS training, including Section 3.4.10, "RCS [reactor coolant system] Pressure and Temperature (P/T) Limits," and simulator training of plant heatup and cooldown, including performance of surveillance tests which ensure P/T curve compliance.

The licensee's outage work control processes, including performance of shutdown risk assessments, provide an additional level of oversight for any activity that could potentially affect reactor level or temperature control during outages.

On the basis of the evaluation of high-pressure injection sources, operator training and established plant-specific procedures, the licensee determined that appropriate controls are in place to minimize the potential for RPV cold over-pressurization events.

3.1.8 Licensee’s Proposed Alternative Examinations

Section 50.55a(a)(3) allows licensees to propose alternatives to the requirements of 10 CFR 50.55a(g). The licensee proposed, as an alternative, to perform examination of essentially 100 percent of the vertical welds and incidental examination of 2 to 3 percent of the intersecting circumferential shell welds. The licensee would permanently defer examination of the circumferential welds until expiration of the plant’s current operating license.

3.1.9 Staff Evaluation of Relief Request RR-22

The NRC staff's review focused on confirming that the licensee has adequately documented that the conditions for relief outlined in the SE to the BWRVIP-05 Report and GL 98-05 are satisfied.

The NRC staff’s SE on the BWRVIP-05 Report, issued July 28, 1998, provided a limiting conditional failure probability of 2×10^{-7} per reactor year for a mean RT_{NDT} at the RPV clad-to-base metal interface of 44.5 °F for CB&I-fabricated RPVs. Comparing the information in the NRC Reactor Vessel Integrity Database (RVID) with that submitted in the licensee’s relief request, the NRC staff has determined that the mean RT_{NDT} of the circumferential welds at SSES Units 1 and 2 is projected to be -30 °F and 10 °F, respectively, at the end of the current license (EOL). In this evaluation, the chemistry factor, ΔRT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2. The difference between the RVID mean RT_{NDT} of 10 °F and the licensee's projected value of 4.9 °F is due to the licensee's use of the projected fluence at the RPV 1/4-T depth at EOL to determine ΔRT_{NDT}, whereas the RVID value for ΔRT_{NDT} was calculated using the projected fluence at the RPV clad-to-base metal interface at EOL. Demonstration of an acceptable EOL mean RT_{NDT} by calculation using the clad-to-base metal interface fluence is required to maintain consistency between the technical basis cited in the NRC staff’s BWRVIP-05 SE and the plant-specific calculations. The calculated values of mean RT_{NDT} at the RPV clad-to-base metal interface for the circumferential welds at SSES Units 1 and 2 are significantly lower than that for the limiting plant-specific case for CB&I-fabricated RPVs, indicating that the conditional failure probability of the SSES Units 1 and 2 circumferential welds is much less than 2×10^{-7} per reactor year.

The NRC staff concludes that a non-design basis cold over-pressure transient is unlikely to occur at SSES. The information provided regarding the SSES high-pressure injection systems, operator training, and plant-specific procedures provides a sufficient basis to support approval of the alternative examination request.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's submittal and finds that the licensee has provided an acceptable demonstration that the appropriate criteria in GL 98-05 and the NRC staff's evaluation of the BWRVIP-05 Report have been satisfied regarding permanent relief (i.e., for the remaining term of operation under the initial, existing license) from ISI requirements for the volumetric examination of reactor pressure vessel circumferential welds, ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

The NRC staff concludes that authorization of the licensee's alternative examinations would provide assurance of structural integrity and, therefore, an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination for SSES Units 1 and 2 is authorized.

Principal Contributors: R. Schaaf
C. E. Carpenter

Date: February 28, 2001