September 14, 2000

Mr. Ronald DeGregorio
Vice President Oyster Creek
AmerGen Energy Company, LLC
P.O. Box 388
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SUBJECT: SAFETY EVALUATION OF THE REQUEST FOR RELIEF FROM THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (ASME CODE) SECTION XI REQUIREMENTS FOR THE CONTAINMENT INSERVICE INSPECTION PROGRAM, OYSTER CREEK NUCLEAR GENERATING STATION (TAC NO. MA8016)

Dear Mr. DeGregorio:

By letter dated December 30, 1999, you submitted Relief Request R-18 concerning the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI requirements for the Oyster Creek Nuclear Generating Station Containment Inservice Inspection (ISI) Program. You requested approval for the use of alternative inspection to perform circumferential shell weld examinations on the reactor pressure vessel (RPV) welds. These examinations are required by the augmented examination requirements of 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 by 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, 1985, and the staff's evaluation of the BWRVIP-05 report issued July 28, 1998. We have reviewed your request, and, based on the information provided, we conclude that the alternative you have proposed will provide an acceptable level of quality and safety for the third 10-year inservice inspection. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

On the date of the December 30, 1999, application, GPU Nuclear, Inc. (GPUN) was the licensed operator for Oyster Creek. On August 8, 2000, GPUN's ownership interest in Oyster Creek was transferred to AmerGen Energy Company, LLC (AmerGen). By letter dated August 10, 2000, AmerGen requested that the Nuclear Regulatory Commission continue to review and act upon all requests before the Commission which had been submitted by GPUN. Accordingly, the staff has completed its review of the requested relief.

Our detailed evaluation and conclusions are documented in the enclosed safety evaluation.

Sincerely,

/RA by Peter S. Tam for/

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

Docket No. 50-219

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE THIRD 10-YEAR INSERVICE INSPECTIONPROGRAM

RELIEF REQUEST R-18

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated December 30, 1999, GPU Nuclear, Inc. (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. 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 Section 50.55a(g)(6)(ii)(A)(2) to Title 10 of the Code of Federal Regulations (10 CFR 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 staff's evaluation of the BWRVIP-05 Report issued July 28, 1998.

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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.

3.0 EVALUATION

3.1 Relief Request No. 18 - Alternatives for Examination of Reactor Pressure Vessel Shell Welds

3.1.1 Code Requirements

10 CFR 50.55a(g)(6)(ii)(A) 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. 10 CFR 50.55a(a)(3) 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 of the BWRVIP-05 report. This evaluation concluded that the failure frequency of RPV circumferential welds in boiling-water reactors (BWRs) was sufficiently low to justify elimination of inservice inspection of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation indicated that examination of the circumferential 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 staff 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. 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 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 (CE), and Babcock and Wilcox (B&W). 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 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 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.

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 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 inservice inspection 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 safety evaluation 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 safety evaluation dated July 28, 1998.

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

3.1.2 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: NR02-1-563, NR02-1-572, NR02-3-572, and NR02-3-564.

3.1.3 Licensee’s Basis for Relief

The licensee's request is based upon provisions in the NRC Safety Evaluation (SE) for the BWRVIP-05 Report and the guidance outlined in GL 98-05. These documents provide the basis for the elimination of inservice inspections 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 established 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.

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 Combustion Engineering (CE), using the highest mean irradiated RT_{NDT} to determine the limiting case.

Since the Oyster Creek RPV was fabricated by CE, the relief request compared the mean RT_{NDT} at 32 effective full-power years (EFPYs) for Oyster Creek to that for the limiting CE case described in Table 2.6-4 of the Final SE of the BWRVIP-05 Report. As illustrated in Table 1, the mean RT_{NDT} for Oyster Creek is lower than that for the limiting CE case, and the licensee concluded that the conditional failure probability for the Oyster Creek circumferential welds is bounded by the conditional failure probabilities in the staff’s SE report through the end of the current license period.

Table 1: Comparison of Oyster Creek Circumferential Weld and the USNRC Limiting Plant Specific Analysis from Table 2.6-4 of the Final SE of the BWRVIP-05 Report

PARAMETER	OYSTER CREEK COMPARATIVE DATA AT 32 EFPY (BOUNDING CIRC. WELD-HEAT 1248)	USNRC LIMITING PLANT SPECIFIC ANALYSIS DATA AT 32 EFPY (CE "CEOG")
Fluence (10^{19} n/cm^2)	0.374	0.2
Initial RT_{NDT} (°F)	-50	0
Chemistry Factor (°F)	98	172.2
Cu (Wt. %)	0.21	0.183
Ni (Wt. %)	0.07	0.704
ΔRT_{NDT} (°F)	71	98.1
Mean RT_{NDT} (°F) [Initial RT_{NDT} + ΔRT_{NDT}]	77	98.1

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

During review of the BWRVIP-05 Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," the 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 assessed the systems that could lead to a cold over-pressurization of the Oyster Creek RPV. These include the standby liquid control (SLC), control rod drive (CRD), and the condensate/feedwater systems. Oyster Creek has no high-pressure core injection system.

There are no automatic starts associated with the SLC system. SLC injection requires operator action to manually start the system from the control room using a key lock controller. In addition, in the event of manual initiation during shutdown, the SLC injection rate of approximately 30 gpm would allow operators sufficient time to control reactor pressure.

During normal cold shutdown conditions, RPV level and pressure are normally controlled through a feed and bleed process using the CRD and Reactor Water Cleanup (RWCU) systems. Plant procedures are in place to respond to any unexpected or unexplained rise in reactor water level.

The CRD system and the RWCU system are also used to control RPV level and pressure during pressure testing of the RPV. The rate of pressure increase is limited to less than 50 psig per minute. This practice minimizes the likelihood of exceeding the pressure-temperature limits during the test.

During normal cold shutdown conditions, the feedwater pumps are secured. Several operator errors would have to occur to result in inadvertent injection into the vessel. One condensate pump is left operating on minimum recirculation flow. The operating condensate pump maintains the system full and can be used as a source of makeup to the RPV. The shutoff head of the pump is approximately 350 psig.

In all cases, the operators are trained in methods of controlling water level within specified limits in addition to responding to abnormal water level conditions during shutdown. Procedures and controls for reactor temperature, level, and pressure are in place to minimize the potential for RPV cold over-pressure events. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits.

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

3.1.4 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 alterative, to perform vertical weld examinations and incidental examination of 2 to 3 percent of the intersecting circumferential shell welds to the maximum extent possible based on accessibility. The licensee would permanently defer examination of the circumferential welds until expiration of the plant's current operating license.

3.1.5 Staff Evaluation of Relief Request R-18

The 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 staff's SE provides a limiting conditional failure probability of 6.34×10^{-5} per reactor year for a limiting plant-specific mean RT_{NDT} of 98.1 °F for CE-fabricated RPVs. Comparing the information in the NRC Reactor Vessel Integrity Database (RVID) with that submitted in the relief request, the staff has confirmed that the mean RT_{NDT} of the circumferential welds at Oyster Creek is projected to be 77 °F at the end of the current license. 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 calculated value of mean RT_{NDT} for the circumferential welds at Oyster Creek is significantly lower than that for the limiting plant-specific case for CE-fabricated RPVs, indicating that the conditional failure probability of the Oyster Creek circumferential welds is much less than 6.34×10^{-5} per reactor year.

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 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 inservice inspection 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 Oyster Creek is authorized.

Principal Contributors: M. Khanna
A. Cubbage

Date: September 14, 2000