April 11, 2003

Mr. Mike Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: PILGRIM NUCLEAR POWER STATION - PILGRIM RELIEF REQUEST NO. 28, RELIEF FROM ASME CODE, SECTION XI, EXAMINATIONS OF REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELDS (TAC NO. MB6074)

Dear Mr. Bellamy:

By letter dated July 1, 2002, Entergy Nuclear Operations, Inc. (ENO, the licensee), submitted Pilgrim Relief Request (PRR)-28 requesting relief from American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements related to examination of reactor pressure vessel circumferential shell welds at Pilgrim Nuclear Power Station. In response to the U.S. Nuclear Regulatory Commission staff's (NRC or the staff) request for additional information dated January 22, 2003, the licensee revised PRR-28 by letter dated February 3, 2003. The revised submittal, PRR-28, Revision 1, replaced the licensee's original request. The licensee submitted the request pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i) and consistent with NRC 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," as a proposed alternative to the requirements of ASME Code, Section XI.

The staff has reviewed the request against the requirements of ASME Code, Section XI, 1989 Edition, Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item No. B1.11, the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2), and the guidance of GL 98-05. The results of the review are documented in the enclosed Safety Evaluation.

The licensee's letter requested relief from the ASME Code, Section XI requirements pursuant to 10 CFR 50.55a(a)(3)(i). However, the applicable requirements for proposing alternatives related to augmented reactor shell weld examinations are contained in 10 CFR 50.55a(g)(6)(ii)(A)(5). The 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 portion of the initial license period that expires in 2012. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5).

If you have any questions, please contact Travis Tate at (301) 415-8474.

Sincerely,

/RA by JBoska for/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVES FOR EXAMINATION OF REACTOR PRESSURE VESSEL SHELL WELDS

PILGRIM NUCLEAR POWER STATION

ENTERGY NUCLEAR GENERATION COMPANY

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated July 1, 2002, Entergy Nuclear Operations, Inc. (ENO, the licensee), submitted Pilgrim Relief Request (PRR)-28 requesting relief from American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, requirements related to examination of reactor pressure vessel (RPV) circumferential shell welds at Pilgrim Nuclear Power Station (Pilgrim). In response to the U.S. Nuclear Regulatory Commission (NRC) staff's request for additional information dated January 22, 2003, the licensee revised PRR-28 by letter dated February 3, 2003. The revised submittal, PRR-28, Revision 1, replaced the licensee's original request.

The relief request would authorize the use of a proposed alternative to the RPV shell welds examination requirements of ASME Code, Section XI, for the remaining portion of the initial license period that expires in 2012.

2.0 REGULATORY EVALUATION

2.1 Applicable Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(4), requires that ASME Code Class 1, 2, and 3 components must meet the requirements set forth in 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.[1] The regulations require that all inservice examinations 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 ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 10-year interval. For Pilgrim, the 1989 Edition to ASME Code, Section XI, is the applicable edition for the current 10-year inservice inspection (ISI) interval.

1. Except for design and access provisions and preservice inspection requirements.

Enclosure

10 CFR 50.55a(g)(6)(ii)(A)(2) requires that all licensees augment their reactor vessel examination by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item No. B1.10, Examination Category B-A, “Pressure Retaining Welds in Reactor Vessel,” Table IWB-2500-1 to Section XI. Additionally, 10 CFR 50.55a(g)(6)(ii)(A)(2) requires that the examinations cover essentially 100 percent of the RPV shell welds. Both Examination Category B-A and 10 CFR 50.55a(g)(6)(ii)(A)(2) define "essentially 100-percent” examination as covering 90 percent or more of the examination volume of each weld. The schedule for implementation of the augmented inspection is dependent upon the number of months remaining in the 10-year ISI interval that was in effect on September 8, 1992.

Alternatives to requirements may be authorized or relief granted by the NRC pursuant to 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), 10 CFR 50.55a(g)(6)(i) or 10 CFR 50.55a(g)(6)(ii)(A)(5). In proposing alternatives or requesting relief from the augmented reactor shell weld examinations under 50.55a(g)(6)(ii)(A), the licensee must demonstrate that the proposed alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(g)(4)(iv), ISI items may meet the requirements set forth in subsequent editions and addenda of the ASME Code that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein, and subject to Commission approval. Portions of editions and addenda may be used provided that related requirements of the respective editions and addenda are met.

The licensee submitted the request pursuant to 10 CFR 50.55a(a)(3)(i) and consistent with NRC Generic Letter (GL) 98-05, “Boiling Water Reactor (BWR) Licensees Use of the BWR Vessel and Internals Project (BWRVIP)-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds,” dated November 10, 1998, as a proposed alternative to the requirements of ASME Code, Section XI. However, the applicable requirements for proposing alternatives related to augmented reactor vessel shell weld examinations are contained in 10 CFR 50.55a(g)(6)(ii)(A)(5). The licensee’s proposed alternative is also consistent with the staff’s safety evaluation report (SER), on EPRI Technical Report TR-105697, “Boiling Water Reactor Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),” issued July 28, 1998.

2.2 BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29,1996, and May 16, June 4, June 13 and December 18, 1997, and January 13, 1998, the BWRVIP, submitted the proprietary report BWRVIP-05. 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 approximately 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 an SER on the BWRVIP-05 report. This evaluation concluded that the failure frequency of RPV circumferential welds in BWRs was sufficiently low to justify elimination of 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 indicated that examination of the circumferential welds shall be performed if axial weld examinations reveal 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 that 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'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 respectively in Tables 2.6-4 and 2.6-5 of the staff's July 28, 1998, SER. 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 SER 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.

2.3 GL 98-05

On November 10, 1998, the NRC issued GL 98-05 which 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 No. B1.11, "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 July 28, 1998, safety evaluation, 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 July 28, 1998, safety evaluation.

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

3.0 TECHNICAL EVALUATION

3.1 Code Requirement for which Relief is Requested

The licensee requested relief from the requirements of ASME Code, Section XI, 1989 Edition, Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item No. B1.11. In addition, the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2) requires volumetric examination of essentially 100 percent of RPV circumferential weld and base material regions in the RPV each inspection interval.

3.1.1 Component(s) for which Relief is Requested

The requested 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"

3.2 Licensee's Proposed Alternative to the ASME Code

Pursuant to the alternative provisions in 10 CFR 50.55a(a)(3)(i), and in compliance with the GL 98-05 provisions in the BWRVIP-05 safety evaluation, in its February 3, 2003 letter, the licensee proposed the following for the remaining portion of the initial operating license period that expires in 2012:

> All longitudinal (axial) RPV shell welds, Code Item B1.11, will be examined to the extent possible. If less than 90% coverage is achieved we shall request additional relief. In addition, essentially 1 to 3% of the incidental intersecting circumferential welds will be examined during refueling outage (RFO)-14 and/or RFO-15 (Third ISI interval).

3.3 Licensee's Bases for Alternative

The licensee states:

> Pursuant to 10 CFR 55.55(a)(3)(i), and consistent with information contained in NRC Generic Letter 98-05, Pilgrim is requesting an alternative from ASME [Code,] Section XI requirements to examine essentially 100% of accessible Category B-A circumferential welds and is proposing permanent relief (for the remaining portion of the initial license period) from these examinations.
>
> Pilgrim completed the first augmented RPV shell weld examination in 1995 during RFO 10. Refer to BECO [Boston Edison Co.] Letter 95-099. This examination included both horizontal and vertical RPV shell weld[s] to the extent possible and detected no flaws within the criteria of ASME [Code, Section] XI[,] IWB-3500.
>
> Consistent with the NRC Generic Letter 98-05[,] the following is provided.
>
> 1. At expiration of license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staffs July 30, 1998 safety [evaluation]. The NRC evaluation of BWRVIP-05 utilized

the FAVOR code to perform a probabilistic fracture mechanics analysis to estimate RPV failure probabilities. Although BWRVIP-05 provides the technical basis supporting the relief request, the following information is provided that shows Pilgrim['s] vessel is enveloped by the NRC analysis.

Pilgrim RPV Shell Weld Information
Bounding Circumferential Weld (1-344)
Wire Heat/Lot (21935/3869)

Parameter Description	Pilgrim RPV Shell Bounding Beltline 32 EFPY Bounding Comparative Parameters (Bounding Circumferential, Weld)	USNRC Limiting 32 EFPY Bounding CE Vessel Parameters SER Table 2.6-4
Neutron fluence at the end of the requested relief period (upper bound value @ 1/4 t)	8.03×10^{17} n/cm^2	2.0×10^{18} n/cm^2
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-50	0
Weld Chemistry Factor (CF), °F	172.2	172.2
Weld Copper content %	0.183	0.183
Weld Nickel content %	0.704	0.704
Increase in reference temperature (ΔRT_{NDT}), °F	64.5	98.1
Mean adjusted reference temperature (ART), °F $RT_{NDT(u)} + \Delta RT_{NDT}$	14.5	98.1

2. Pilgrim has implemented Operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staffs July 30, 1998 safety evaluation.

3.4 Evaluation

As described previously, GL 98-05 provides two criteria that BWR licensees requesting relief from ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No.B1.11, Circumferential Shell Welds) must satisfy. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the safety evaluation. The staff has reviewed the licensee's regulatory and technical analyses described in Attachments 1 and 2 of the February 3, 2003, submittal to determine if the licensee's proposed alternative inspection provides an acceptable level of quality and safety.

3.4.1 Circumferential Weld Conditional Failure Probability

The staff's SER 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 CE, using the highest mean irradiated RT_{NDT} to determine the limiting case. Since the Pilgrim RPV was fabricated by CE, the licensee compared the mean irradiated RT_{NDT} for Pilgrim to that for the limiting CE case described in Table 2.6-4 of the staff's SER for the BWRVIP-05 report. As indicated in the licensee's evaluation, the mean RT_{NDT} for Pilgrim is lower than that for the limiting CE case; therefore, the licensee concluded that the conditional failure probability for the Pilgrim circumferential welds is bounded by the conditional failure probabilities in the staff's SER through the end of the current license period.

The staff's SER 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 with that submitted in the proposed relief request, the staff confirmed that the mean RT_{NDT} of the circumferential welds at Pilgrim is projected to be 14.5 °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 (RG) 1.99, Revision 2. The calculated value of mean RT_{NDT} for the circumferential welds at Pilgrim is significantly lower than that for the limiting plant-specific case for CE-fabricated RPVs, indicating that the conditional failure probability of the Pilgrim circumferential welds is much less than 6.34×10^{-5} per-reactor-year.

The regulatory requirements for fluence calculations are defined in General Design Criteria (GDCs) 30 and 31, as established in Appendix A to 10 CFR Part 50. Additionally, in March 2001, the staff augmented those GDCs by issuing RG 1.190. As a result, when issuing License Amendment No. 190, dated April 13, 2001 (ML011060046), the staff noted that Pilgrim's plant-specific calculations for the original fluence value were outdated. The staff concluded that fluence calculations are acceptable if they are done with approved methodologies or other methods that are shown to adhere to the guidance of RG 1.190.

In regard to this relief request, the staff also evaluated the impact of the licensee's projected fluence value on Pilgrim's circumferential weld conditional failure probability since the calculation does not comply with the guidance of RG 1.190. The staff verified the licensee's input parameters and performed fluence calculations using the limiting CE vessel mean RT_{NDT}. The staff determined that the licensee's plant-specific calculations for the fluence value would have to be a factor of 10 higher to exceed the limiting case for CE-fabricated RPVs. Although the staff believes the licensee's fluence value calculation is outdated, the staff does not believe performing the fluence calculation in accordance with RG 1.190 would increase the licencee's estimate by a factor of 10. Therefore, the staff concludes that the licensee's fluence calculation does not significantly impact the calculated value of mean RT_{NDT} for the circumferential welds at Pilgrim and that the mean RT_{NDT} would remain below the limiting case for CE-fabricated vessels.

3.4.2 Cold Over-pressure Transient Probability

To satisfy the second condition of GL 98-05 regarding a cold over-pressure event, the licensee provided its analysis of the high-pressure injection sources, administrative controls, and operator training.

High-Pressure Injection Sources

The licensee assessed the systems that could lead to a cold over-pressurization of the Pilgrim RPV. These include the high-pressure core injection (HPCI), the reactor core isolation cooling (RCIC) system, the feedwater system, the condensate system, the standby liquid control (SLC) system, the low-pressure coolant injection (LPCI) system, the core spray (CS) system, the residual heat removal (RHR) system, the control rod drive (CRD) system, and the reactor water clean-up (RWCU) system. The licensee stated that the HPCI and RCIC systems use steam driven turbines to pump cold water into the vessel. During cold shutdown conditions, no steam is available to operate these systems; therefore, they could not cause a cold over-pressure event.

The feedwater/condensate systems are a potential source of high-pressure injection into the reactor vessel. The condensate pumps are a source of water to the reactor feed pumps. The reactor feed pumps are the high pressure makeup system during normal operation. The licensee's evaluation considered the availability of automatic trips of the condensate/feedwater systems on high vessel water level. Additionally, the licensee indicated that during refueling outages the feedwater lines are isolated by closing block valves inside the drywell. The feedwater regulating valves are also removed from service once the plant is at low power (approximately 10 percent). Controls on pressure and temperature are imposed by startup and cool down procedures which limit the capacity of the condensate/feedwater systems to inject water with reactor coolant temperature below 450 °F. The licensee concluded the potential for over-pressurization associated with feedwater/condensate systems is not significant.

The SLC system is an additional high-pressure source. However, there are no automatic starts associated with the SLC system. The system requires operator action to manually start the system by a key-lock switch; therefore, inadvertent manual initiation of SLC is an unlikely event. Additionally, in the event of manual initiation during shutdown, the maximum SLC injection rate of approximately 40 gpm would allow operators sufficient time to control reactor pressure. Procedures have been developed for operation of the SLC system and operators are trained on the system operation. The licensee concluded that the SLC does not present a significant potential for over-pressurization.

The licensee indicated that the LPCI, CS, and RHR systems operate at pressures below 400 psig. The pressure-temperature (P-T) limits permit pressures up to 310 psig at temperatures from 70 °F to 100 °F. At 100 °F, the permitted pressure increases rapidly to 660 psig. The shutoff head pressure for a CS pump is nominally 350 psig and a RHR pump is nominally 332 psig. Since these systems could pressurized the RPV to greater than 310 psig, the licensee evaluated the potential for over-pressurization associated with the LPCI, RHR, and CS operating modes.

During emergency situations, following a loss of shutdown cooling, an alternate shutdown cooling mode is permitted which uses a CS pump to circulate water from the condensate storage tanks through the reactor vessel and safety relief valve (SRV) discharge lines to the torus. In this mode, the SRV control solenoids are energized allowing the SRV main discharge to open at 50 psig and coolant to exit the reactor vessel and flow to the torus. This mode of operation is permitted only when the P-T limits can be maintained and prevents the reactor pressure from exceeding 310 psig. Interlocks are in place to prevent flow into the reactor vessel during normal testing of the LPCI and CS with reactor pressure above 400 psig. During testing under shutdown conditions, an inadvertent injection of the LPCI or CS would be detected by operators and alarm of the reactor vessel level annunciator before exceeding 310 psig, and the injection would be terminated. Reactor head vents are administratively controlled to remain open until the reactor coolant temperature is greater than 212 °F and the vessel metal temperatures are acceptable for critical core operations. The reactor head vents prevent reactor vessel pressurization above 310 psig until operator action when using the RHR or CS pumps for reactor vessel fill from a cold and vented condition. Reactor head vents also prevent over-pressurization when using a CS pump for reactor vessel and cavity fill during an outage because the head vents open when the vessel is filled to the flange. Procedures are in place to limit injection during full-flow testing of the CS injection check valves under shutdown conditions. The procedures instruct operators to minimize reactor vessel level rise and the injection duration is limited to that which indicates a reactor vessel level change. These administrative actions prevent reactor pressure from exceeding 310 psig. While utilizing the shutdown-cooling mode of the RHR system, design interlocks close the suction valves from the reactor when pressure is greater than 70 psig. Closure of these suction valves will trip the running RHR pump and prevent over-pressurization. The licensee concluded the LPCI, CS, and RHR systems do not present a significant potential for over-pressurization.

During normal cold shutdown conditions, RPV level and pressure are controlled with the CRD system and the RWCU system using a “feed and bleed” process. The licensee has procedures in place to reduce the likelihood of a violation of the P-T limits, such as requiring the opening of the head vent valves after the reactor has been cooled to less than 220 °F. In addition, the slow injection rate of the CRD system (50 gpm) and the RWCU system (222 gpm) under these conditions, would allow the operator sufficient time to react to unanticipated changes in RPV level. During the RPV pressure test, procedural guidance is provided to allow operators to administratively monitor the RPV pressure and temperature while controlling CRD and RWCU injection to assure a smooth, controlled method of increasing or decreasing pressure while the vessel temperature is being maintained above the required P-T limits. The licensee concluded that the CRD and RWCU systems do not present a significant potential for over-pressurization.

The staff has reviewed the licensee’s assessment of systems that could lead to a cold over-pressurization of the RPV. Based on this assessment, the staff determined that the licensee has adequately identified systems that could cause a cold over-pressurization event. In addition, the staff determined that the licensee has implemented operator training and established procedures to limit the potential for a cold over-pressure event associated with the systems. Therefore, the staff concludes that the licensee’s assessment of high-pressure injection sources satisfy the criteria of GL 98-05.

Operating Training and Work Control Process

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

The licensee indicated that, during plant outages, work control procedures ensure that outage schedules and changes to schedules receive a risk assessment review commensurate with their safety significance. Additionally, individuals involved in work activity attend pre-job briefings that include discussions of expected plant responses and contingency actions which address unexpected conditions or responses that may be encountered.

The licensee concluded that the probability of a low temperature RPV over-pressure is bounded by that used in the staff's SE.

The staff has reviewed the licensee's evaluation of a cold over-pressure event. In particular, the staff reviewed the licensee's evaluation of the high-pressure injection sources, and its discussion of the operator training and work control process. Based on the implementation of operator training and establishment of procedures, the staff concludes that the licensee has provided an acceptable demonstration that the criteria in GL 98-05 is satisfied in regard to a cold over-pressure event.

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 portion of the initial license period) from ISI requirements of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, for the volumetric examination of RPV circumferential welds.

The staff concludes that the licensee's proposed alternative examination is authorized pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) for the remaining portion of the initial license period that expires in 2012, on the basis that the alternative provides an acceptable level of quality and safety.

All other requirements of the ASME Code, Section III and XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: S. Wall
T. Tate

Date: April 11, 2003