UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE TO REACTOR PRESSURE VESSEL

CIRCUMFERENTIAL WELD INSPECTIONS

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

PECO ENERGY COMPANY

DOCKET NO. 50-277

1.0 INTRODUCTION

By letter dated April 2, 1998, as supplemented by letter dated August 12, 1998, PECO Energy Company (the licensee) requested an alternative to performing the reactor pressure vessel (RPV) circumferential shell weld examination requirements of both the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1980 Edition, with the Winter 1981 Addenda (inservice inspection (ISI)), and the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2), for the Peach Bottom Atomic Power Station (PBAPS), Unit 2.

Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2 and 3 components must 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 the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code, Section XI, for PBAPS, Unit 2, during the current 10-year ISI interval is the 1980 Edition, through the Winter 1981 Addenda.

Section 50.55a(g)(6)(ii)(A) to Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a(g)(6)(ii)(A)) requires that licensees perform an expanded RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code, on an "expedited" basis. "Expedited," in this context, effectively meant during the inspection interval when the Rule was approved or the first period of the next inspection interval. 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

ENCLOSURE

examinations of "essentially 100 percent" of the RPV pressure-retaining shell welds during all inspection intervals. Section 50.55a(a)(3)(i)(10 CFR 50.55a(a)(3)(i)) indicates that alternatives to the requirement in 10 CFR 50.55a(g) are justified when the proposed alternative provides an acceptable level of quality and safety.

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, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," which proposed to reduce the scope of inspection of the BWR RPV welds from essentially 100 percent of all RPV shell welds to 50 percent of the longitudinal welds and 0 percent of the circumferential welds. By letter dated October 29, 1996, the BWRVIP modified their proposal to increase the examination of the longitudinal welds to 100 percent from 50 percent while still proposing to inspect essentially 0 percent of the circumferential RPV shell welds, except that the intersection of the longitudinal and circumferential welds would have included approximately 2-3 percent of the circumferential welds.

On May 12, 1997, the NRC staff and members of the BWRVIP met with the Commission to discuss the NRC staff's review of the BWRVIP-05 report. In accordance with guidance provided by the Commission in Staff Requirements Memorandum (SRM) M970512B, dated May 30, 1997, the staff has initiated a broader, risk-informed review of the BWRVIP-05 proposal, and issued a final safety evaluation related to the review of BWRVIP-05 on July 28, 1998, which generically approved the reduction in inspection of circumferential RPV welds.

In IN 97-63, Supplement 1, the staff indicated that it would consider technically justified alternatives to the augmented examination in accordance with 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), and 50.55a(g)(6)(ii)(A)(5), from BWR licensees who are scheduled to perform inspections of the BWR RPV circumferential welds during the fall 1998 or spring 1999 outage seasons. Acceptably justified alternatives would be considered for inspection delays of up to 40 months or two operating cycles (whichever is longer) for BWR RPV circumferential shell welds only.

2.0 BACKGROUND - NRC STAFF ASSESSMENT OF BWRVIP-05 REPORT

The staff's independent assessment of the BWRVIP-05 proposal is documented in a NRC letter dated August 14, 1997, from Mr. Brian Sheron to Mr. Carl Terry, BWRVIP Chairman. The staff concluded that the industry's assessment did not sufficiently address risk, and additional work was necessary to provide a complete risk-informed evaluation. The staff's assessment was performed for BWR RPVs fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The staff assessment identified pressure and temperature resulting from a cold overpressure event in a foreign reactor as the limiting event for BWR RPV's. The materials and neutron radiation parameters used by the staff are identified in Table 7-1 of the staff's independent assessment. The staff determined the conditional probability of failure for longitudinal and circumferential welds fabricated by CB&I, CE and B&W. Table 7-9 of the staff's assessment identifies the conditional probability of failure for the reference cases and the 95% confidence uncertainty bound cases for longitudinal and circumferential welds fabricated by CB&I, CE and B&W. B&W fabricated vessels were determined to have the highest conditional probability of failure. The input material parameters used in the analysis of the reference case for B&W fabricated vessels resulted in a reference temperature (RT_{NDT}) at the vessel inner surface of 114.5 °F. In the uncertainty analysis, the neutron fluence evaluation had the greatest RT_{NDT} value (145 °F) at the inner surface. Vessels with RT_{NDT} values less than those resulting from the staff's assessment will have less embrittlement than the vessels simulated in the staff's assessment and should have a conditional probability of vessel failure less than or equal to the values in the staff's assessment.

The failure probability for a weld is the product of the critical event frequency and the conditional probability of the weld failure for that event. Using the event frequency for a cold overpressure event and the conditional probability of vessel failure for B&W fabricated circumferential welds, the best-estimate failure frequency from the staff's assessment is 6×10^{-10} [(1)] per reactor year and the upper bound failure frequency from the uncertainty analysis is 3.9×10^{-8} [(1)] per reactor year.

3.0 PROPOSED ALTERNATIVE

The licensee proposed to defer the examination of the RPV circumferential shell welds as required by 10 CFR 50.55a(g)(6)(ii)(A)(2), and the inservice inspection requirements for circumferential welds in the ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through Winter 1981 Addenda (Table IWB-2500-1, Examination Category B-A, Item No. B1.11) for two operating cycles and to perform an alternate examination in the interim. The proposed alternative is to examine the RPV longitudinal shell welds to the maximum extent practicable, from the inner diameter, within the constraints of vessel internal constructions, which will result in partial examination (2-3 percent) of the circumferential welds at or near the intersection of the longitudinal and circumferential welds. This would be done in accordance with the BWRVIP-05 and the ASME Code requirements (i.e., one-third of the welds inspected each 40 months of the current 10-year interval).

4.0 LICENSEE'S TECHNICAL JUSTIFICATION

The licensee will be performing an examination of the reactor vessel longitudinal shell welds to the maximum extent practical from the inner diameter, within the constraints of vessel internal restrictions. It should be noted that the current examination plan is designed to provide longitudinal welds coverage; however, incidental coverage will result in an estimated inspection of 2-3 percent of the intersecting circumferential welds.

The basis for the request is documented in the report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," that was transmitted to the NRC in September 1995. The BWRVIP-05 report provides the technical basis

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[(1)] Insufficient or no failures to accurately determine reference case failure probability and sensitivity to flaw size, flaw density and inservice inspection.

for eliminating inspection of BWR RPV circumferential shell welds. The BWRVIP-05 report concludes that the probabilities of failure of the BWR RPV circumferential shell welds are orders of magnitude lower than that of the longitudinal shell welds.

The NRC staff has conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05. This independent NRC assessment utilized the FAVOR Code to perform a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. The key parameters in the PFM analysis are the initial RT_{NDT}, the end of license mean neutron fluence, the mean chemistry (percent copper and nickel) of the welds, and the pressure and temperature of the events being considered. The BWRVIP-05 provides the technical basis supporting the alternative, and the following table illustrates that the PBAPS, Unit 2 plant has additional conservatism in comparison to the NRC's Independent Assessment Fracture Analysis limiting case.

PARAMETER DESCRIPTION	PBAPS, UNIT 2 COMPARATIVE PARAMETERS AT 19 EFPY (Bounding Circ. Weld)	NRC INDEPENDENT ASSESSMENT LIMITING FRACTURE ANALYSIS PARAMETERS
Fluence, n/cm^2	5.2×10^{17}	1.25×10^{18}
Initial RT_{NDT}, °F	-32	-5
Chemistry Factor	82	190
Cu%	0.06	0.287
Ni%	0.97	0.60
ΔRT_{NDT}	20.1	87.9
Margin Term	20.1	62.2
Mean ART	-11.9	82.9
Upper Bound ART	8.1	145.1

The chemistry factor, ΔRT_{NDT}, margin term, mean adjusted reference temperature (ART), and upper bound ART are calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2. Since the upper bound ART for the bounding PBAPS, Unit 2 circumferential weld is less than the value from the NRC Independent Assessment, the licensee concluded that the PBAPS, Unit 2 circumferential welds are bounded by the staff's assessment, thus providing additional assurance that the vessel welds are also bound by the BWRVIP-05 report.

As provided in the following discussion, the licensee has in place procedures which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown which would minimize the likelihood of a cold overpressure event from occurring. Additionally, operator training reinforces these procedures.

The ASME Code Leakage Pressure Test and the ASME Code Hydrostatic Pressure Test procedures, which have been used at PBAPS, have sufficient procedural guidance to prevent a cold, overpressurization event. The leakage pressure test is performed at the conclusion of each refueling outage, while the Hydrostatic Pressure Test is performed once every 10 years. Other pressurizations required for informational leakage inspections are performed in accordance with a procedure similar to the ASME Code test procedures. These pressurizations are infrequently-performed, complex tasks, and the test procedures are considered Plant Evolution/Special Tests. As such, a requirement is included in them for operation's section management to perform a "pre-job briefing" with all essential personnel. This briefing details the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar inhouse or industry operating experiences, the importance of open communications, and, finally, the process in which the test would be aborted if plant systems responded in an adverse manner. Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification pressure-temperature curve. Also, the procedures require the designation of a Test Coordinator for the duration of the test who is a single point of accountability, responsible for the coordination of testing from initiation to closure, and maintaining shift management and line management cognizant of the status of the test.

Additionally, to ensure a controlled, deliberate pressure increase, the rate of a 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 a cold overpressure condition, the high pressure make-up systems, High Pressure Coolant Injection (HPCI) System, and the Reactor Core Isolation Cooling (RCIC) System, as well as the normal feedwater supply (via the Reactor Feedwater Pumps) at PBAPS 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 overpressure event while the unit is in cold shutdown.

In the case of low pressure system initiation, the PBAPS, Unit 2 pressure-temperature limit curves for hydrostatic testing (PBAPS, Unit 2 Technical Specifications Figure 3.4.9-1), permit pressures up to 312 psig at temperatures from 70 °F up to 100 °F. Above 100 °F, the permissible pressure increases immediately to above 560 psig and increases rapidly with increasing temperature. The maximum discharge pressure for the PBAPS Core Spray and Residual Heat Removal Pumps, taking a suction from the Torus at atmospheric pressure, is approximately 373 psig and 360 psig, respectively. Therefore, the potential for an over-pressurization event which would significantly exceed the pressure-temperature limits, due to an inadvertent actuation of these systems, is very low.

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 emergency core cooling system (ECCS) injection, tripping CRD pumps, terminating all other injection sources, and lowering RPV level via the RWCU system.

In addition to procedural barriers, the operator training has been provided also to further reduce the possibility of the occurrence of cold overpressure events. During initial licensed operator training the following topics are covered: brittle fracture and vessel thermal stress; Operational Transient (OT) procedures, including the OT on reactor high level Technical Specification training, including Section 3.4.9, "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 for in the stations' operations manual.

In addition to the above, ongoing review of industry operating plant experiences is conducted to ensure that the licensee's procedures consider the impact of actual events, including cold overpressure events. Appropriate adjustments to the procedure and associated training are then implemented, to preclude similar situations from occurring at PBAPS.

Based upon the above, according to the licensee, the probability of a cold overpressure transient is considered to be less than or equal to that used in the NRC analysis.

5.0 STAFF DISCUSSION

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 overpressure transients should be considered. The licensee assessed the systems that could lead to a cold overpressurization of the PBAPS, Unit 2 RPV. These included the high pressure core spray (HPCS), reactor core isolation cooling, standby liquid control (SLCS), reactor feedwater pumps, low pressure core spray (LPCS), low pressure core injection (LPCI), control rod drive, and reactor water cleanup systems (RWCU).

The HPCS and RCIC pumps are steam driven and do not function during cold shutdown. The reactor feedwater pumps are the high pressure makeup system during normal operations. The reactor feedwater pumps are also steam driven and therefore, cannot be operated during cold shutdown. Generally, there are no automatic starts associated with SLCS. Operator initiation of SLCS should not occur during shutdown, however, the SLCS injection rate of approximately 40 gpm would allow operators sufficient time to control reactor pressure if manual initiation occurred.

The LPCS and LPCI systems are low pressure ECCS systems with low shutoff heads. If one of these systems were either manually or inadvertently initiated during cold shutdown, the resulting reactor pressure and temperature would be below the TS pressure-temperature limits. The CRD and RWCU systems use a feed and bleed process to control RPV level and pressure during normal cold shutdown conditions. Plant procedures are in place to respond to any unexpected or unexplained rise in reactor water level which could result from spurious actuation of an injection system. The procedure actions 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 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. The licensee also stated that the procedure controls for reactor temperature, level, and pressure are an integral part of operator training. 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 pressure limits of the operating systems, operator training, and established plant-specific procedures the licensee determined that a non-design basis cold overpressure transient is unlikely to occur during the requested delay.

The staff finds that the information provided on the PBAPS, Unit 2, high pressure injection systems, operator training, and plant-specific procedure provides a sufficient basis to support approval of the alternative examination request. The staff concludes that the probability of a non-design basis cold overpressure transient is low at PBAPS, Unit 2, during the requested delay.

6.0 STAFF REVIEW OF LICENSEE TECHNICAL JUSTIFICATION

The staff confirmed that the RT_{NDT} values for the circumferential welds at the end of the relief period will be less than the values in the reference case and uncertainty analysis for the B&W fabricated vessels. The RT_{NDT} is a measure of the amount of irradiation embrittlement. Since the RT_{NDT} values are less than the values in the reference case and the uncertainty analysis for B&W fabricated vessels, the PBAPS RPV will have less embrittlement than the reference case and will have a conditional probability of vessel failure less than or equal to that estimated in the staff's assessment.

The staff reviewed the information provided by the licensee regarding the PBAPS high pressure injection systems, operator training, and plant-specific procedures to prevent RPV cold over-pressurization. The information provided sufficient basis to support approval of the alternative examination request. The staff concludes that the probability of a non-design basis cold over-pressure transient occurring at PBAPS during the requested delay is low, which is consistent with the staff's assessment.

7.0 CONCLUSIONS

The licensee has shown and the staff has determined that, based on the materials in the circumferential welds in the PBAPS, Unit 2 RPV, the conditional probability of vessel failure is expected to be less than or equal to that estimated by the staff's conservative assessment.

Additionally, the staff finds that the low high pressure injection potential, operator training, and plant specific procedures provide a basis to conclude that a non-design-basis cold overpressure transient during the requested delay period (two operating cycles) is unlikely to occur.

Accordingly, the staff concludes that the proposed alternative to delay by two operating cycles the examination of circumferential welds at PBAPS, Unit 2 provides an acceptable level of quality and safety by providing assurance of structural integrity for the interim period. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

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