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10 CFR 50.55a

RS-09-009
January 16, 2009

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Request for Proposed Alternative to 10CFR50.55a Examination Requirements for Reactor Pressure Vessel Circumferential Shell Welds Pursuant to NRC Generic Letter 98-05 (Request 4215)

References:

(1) NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998

(2) BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 28, 1995

(3) Letter from U. S. Nuclear Regulatory Commission to C. Terry (BWRVIP), "Final Safety Evaluation of the BWR Vessel and Internals Project BWTVIP-05 Report (TAC No. M93925)," dated July 28, 1998

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), and consistent with NRC Generic Letter 98-05 (Reference 1), Exelon Generation Company, LLC (EGC), hereby requests permanent relief (i.e., for the remaining portion of the initial license period that expires on September 29, 2026) from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of reactor pressure vessel (RPV) circumferential shell welds. The inspection requirement is specified in the 1989 American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11.

Reference 1 states that the NRC has completed review of the Report entitled "BWR Vessel and Internals Project [BWRVIP], BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," and that licensees of boiling water reactors (BWRs) may request permanent relief from the inservice inspection requirements of 10 CFR 50.55a(g), "Inservice inspection requirements," for the volumetric examination of circumferential reactor pressure vessel welds. The NRC indicated that it would consider technically justified requests for permanent relief if the licensee demonstrates that: (1) at the expiration of their current license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC'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's July 28, 1998, safety evaluation. Reference 1 also states that licensees will still need to perform the required inspections of "essentially 100 percent" of all axial welds.

EGC requests approval of an alternative reactor pressure vessel examination for Clinton Power Station (CPS), Unit 1. Approval of this alternative examination is requested by January 16, 2010, in accordance with 10 CFR 50.55a(a)(3)(i) for the remaining term of the CPS Unit 1 operating license.

The basis for this alternative inspection is discussed in the attached relief request. The basis is consistent with the methodology used in Reference 2, and the provisions of the NRC's safety evaluation for BWRVIP-05 (i.e., Reference 3). In addition, the alternative inspection meets the aforementioned criteria of Reference 1. EGC has concluded that this alternative inspection provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

There are no regulatory commitments contained in this letter. If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

Respectfully,

Jeffrey A. Hansen
Manager – Licensing
Exelon Generation Company, LLC

Attachment: 10 CFR 50.55a Request Number 4215

10 CFR 50.55a Request Number 4215

1. ASME Code Component(s) Affected

Code Class:	1
Component Numbers:	RPV-C1, RPV-C2, RPV-C3, and RPV-C4
Examination Category:	B-A
Item Number:	B1.11
Description:	Reactor Pressure Vessel (RPV) Shell Circumferential Welds

2. Applicable Code Edition and Addenda

Clinton Power Station (CPS) is currently in its second 10-year inspection interval and complies with the 1989 Edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI. Additionally, for ultrasonic examinations, Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 1995 Edition, with the 1996 Addenda, is implemented as required (and modified) by 10 CFR 50.55a.

3. Applicable Code Requirement

In accordance with the provisions of 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC) requests permanent relief (for the remaining portion of the initial license period that expires on September 29, 2026) for CPS, Unit 1, from the following requirements:

1. Subarticle IWB-2500 requires components specified in Table IWB-2500-1 to be examined. Table IWB-2500-1 requires volumetric examination of all RPV shell circumferential welds each inspection interval (i.e., Examination Category B-A, Item No. B1.11);

2. Subsubarticle IWB-2420 requires the sequence of component examinations which was established during the first inspection interval to be repeated during each successive inspection interval, to the extent practical. Therefore, performance of successive examinations of RPV shell circumferential welds is required by Subsubarticle IWB-2420; and

3. Subsubarticle IWB-2430 requires examinations performed in accordance with Table IWB-2500-1 that reveal flaws or relevant conditions exceeding the acceptance standards of Table IWB-3410-1 to be extended to include additional examinations during the current outage.

4. Reason for Request

Reference 1 provides the technical basis for permanently deferring the augmented inspections of circumferential welds in boiling water reactor (BWR) RPVs. In the report, the BWR Vessel and Internals Project (BWRVIP) concluded that the probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. The NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment (PFMA) of the analysis presented in Reference 1, and the results are documented in Reference 2. EGC has determined that the proposed alternative described below provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

5. Proposed Alternative and Basis for Use

<u>Proposed Alternative</u>
In accordance with 10 CFR 50.55a(a)(3)(i), and consistent with information contained in Reference 3, EGC considers the following alternate provisions for the subject weld examinations.

Inservice Inspection Scope

The failure frequency for RPV shell circumferential welds is sufficiently low to justify their elimination from the ISI requirement of 10 CFR 50.55a(g) based on the NRC Safety Evaluation (Reference 2).

The ISI and augmented examination requirements of the ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, RPV shell longitudinal welds (i.e., also known as vertical or axial welds) shall be performed, to the extent possible, and shall include inspection of the circumferential welds only at the intersection of these welds with the longitudinal welds, or approximately 2 to 3 percent of the RPV shell circumferential welds. When this examination is performed, an automated ultrasonic inspection system will provide the best possible examination of the RPV shell longitudinal welds.

The procedures for these examinations shall be qualified such that flaws relevant to the RPV integrity can be reliably detected and sized, and the personnel implementing these procedures shall be qualified in the use of these procedures.

Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, RPV shell circumferential welds (i.e., at intersections with longitudinal welds), successive examinations per Subsubarticle IWB-2420 are not required for non-threatening flaws (i.e., original vessel material or fabrication flaws such as inclusions which exhibit negligible or no growth during the life of the vessel), provided that the following conditions are met:

1. The flaw is characterized as subsurface in accordance with BWRVIP-05 (i.e., Reference 1);

2. The non-destructive examination technique and evaluation that detected and characterized the flaw as originating from material manufacture or vessel fabrication is documented in a flaw evaluation report; and

3. The vessel containing the flaw is acceptable for continued service in accordance with Subarticle IWB-3600, "Analytical Evaluation of Flaws," and the flaw is demonstrated acceptable for the intended service life of the vessel.

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, RPV shell longitudinal welds, all flaws shall be reinspected at successive intervals consistent with ASME Code and regulatory requirements.

Additional Examinations of Flaws

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, RPV shell circumferential welds (i.e., at intersections with longitudinal welds), additional requirements per Subsubarticle IWB-2430, "Additional Examinations," are not required for flaws provided the following conditions are met:

1. If the flaw is characterized as subsurface in accordance with BWRVIP-05, then no additional examinations are required;

2. If the flaw is not characterized as subsurface in accordance with BWRVIP-05, then an engineering evaluation shall be performed, addressing the following as a minimum:

    - A determination of the root cause of the flaw,
    - An evaluation of any potential failure mechanisms,
    - An evaluation of service conditions which could cause subsequent failure, and
    - An evaluation per Subarticle IWB-3600 demonstrating that the vessel is acceptable for continued service; and

3. If the flaw meets the criteria of Subarticle IWB-3600 for the intended service life of the vessel, then additional examinations may be limited to those welds subject to the root cause conditions and failure mechanisms, up to the number of examinations required by paragraph (a) of Subsubarticle IWB-2430. If the engineering evaluation determines that there are no additional welds subject to the same root cause conditions or no failure mechanism exists, then no additional examinations are required.

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, RPV shell longitudinal welds, additional examination for flaws shall be in accordance with Subsubarticle IWB-2430. All flaws in RPV shell longitudinal welds shall require additional weld examinations consistent with ASME Code and regulatory requirements. Examinations of the RPV shell circumferential welds shall be performed if RPV longitudinal welds reveal an active, mechanistic mode of degradation.

<u>*Basis for Use*</u>

Reference 1 provides the technical basis to justify relief from the examination requirements of RPV shell circumferential welds. The results of the NRC's evaluation of Reference 1 are documented in Reference 2. Reference 3 permits BWR licensees to request permanent relief from the ISI requirements of 10 CFR 50.55a(g) (i.e., for the remaining term of operation under the existing, initial license) for the volumetric examination of RPV shell circumferential welds (i.e., ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11). This relief can be granted by demonstrating that:

1. At the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation, 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 staff's July 28, 1998, safety evaluation.

Reference 3 also states that licensees will still need to perform the required inspections of "essentially 100 percent" of all axial welds.

Generic Letter 98-05, Criterion 1

Demonstrate that at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC's July 28, 1998, safety evaluation.

Response

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a PFMA to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFMA are: (1) the neutron fluence used was the estimated end-of-life mean fluence, (2) the chemistry values are mean values based on vessel types, and (3) the potential for beyond-design-basis events is considered.

Table 1 provides a comparison of the limiting RPV circumferential weld parameters for CPS to those found in Table 2.6-4 of the NRC final safety evaluation of BWRVIP-05 (i.e., Reference 2) for a Chicago Bridge and Iron (CB&I) vessel. The material composition and chemistry factors, and the inside diameter fluences at 32 effective full power years

(EFPYs) were used to determine the acceptable reference temperatures at CPS. Although the unirradiated reference temperature for CPS is higher than the NRC limit, the combination of unirradiated reference temperature and embrittlement shift yields adjusted reference temperatures considerably lower than the NRC mean analysis values.

As a result, the shift in reference temperature is lower than the 32 EFPY shift from the NRC analysis. Therefore, the RPV shell weld embrittlement due to fluence is calculated to be less than the NRC's limiting case, and the RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability in the NRC's limiting plant specific analysis (32 EFPY) through the projected end of license. For these reasons, the limiting conditional failure probability for CPS RPV circumferential welds is bounded by Reference 2.

Table 1
Effects of Irradiation on RPV Circumferential Weld Properties

Parameter Description	CPS Parameters at 32 EFPY (Weld Wire Heat/Flux Lot #76492/L430B27AE)	NRC Limiting Plant Specific Analysis*
Copper (weight %)	0.10	0.10
Nickel (weight %)	1.08	0.99
Chemistry Factor	135	134.9
End of Life Inside Diameter Fluence (10^{19} n/cm^2)	0.081	0.51
ΔRT_{NDT} (^{o}F)	50.77	109.5
$\Delta RT_{NDT(U)}$ (^{o}F)	-30	-65
Mean RT_{NDT} (^{o}F)	20.77	44.5

*** Table 2.6-4, "Summary of Results of NRC Staff and BWRVIP Limiting Plant-Specific Analyses (32 EFPY)," corrected per Reference 8.**

Generic Letter 98-05, Criterion 2

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

Response

Procedures are in place for CPS that guide operators in controlling and monitoring reactor pressure during all phases of operation, including cold shutdown. Use of these procedures will prevent an over-pressure event, and are reinforced through operator training. Operating procedures contain sufficient guidance to prevent a low temperature over-pressurization event. A reactor coolant system leakage test is performed prior to each restart after a refueling outage. A pre-job briefing is required prior to test commencement with all involved personnel. During pressure testing, measures are taken to limit the potential for system perturbations that could lead to pressure transients. These measures include both administrative and/or hardware controls, such as limiting testing or work activities, or installing jumpers or simulators, to defeat systems actuations that are not required to be operable. Vessel temperature and pressure are required to be monitored and controlled to within CPS Technical Specifications pressure and temperature (P/T) limits during all portions of testing. Pre-job briefings and careful coordination ensure that pressure transients are minimized.

The high pressure coolant sources that could inadvertently initiate and result in a low temperature overpressurization event are the Feedwater, Reactor Core Isolation Cooling (RCIC), and High Pressure Core Spray (HPCS) systems. During normal RPV fill prior to pressure testing, the Control Rod Drive (CRD) system is the preferred method for filling the reactor. The Condensate/Condensate Booster systems are used as an alternative means to fill the reactor. The motor driven reactor feedwater pump is prevented from starting by the high water level feedwater pump trip signal, which is present due to the high reactor water levels required during pressure testing. During the reactor coolant system leakage test, the reactor is in cold shutdown, and as a result, there is no steam available to drive the turbine driven RCIC and turbine driven reactor feedwater pumps.

The HPCS system is a high pressure make-up system at CPS. The HPCS pump is motor operated, so it can be operated when the reactor is in cold shutdown. However, the HPCS system would require manual initiation, inadvertent initiation, or manual startup to start and inject into the RPV. Also, there is a high RPV water level interlock for the HPCS injection valve to prevent overfilling the RPV. This high level interlock is not normally overridden. Even if the HPCS system is inadvertently started, it would not inject and pressurize the reactor due to the high RPV water level interlock.

The CRD system is a high pressure system used to operate the control rods. The CRD system is a low flow rate system with about 50 gpm flow rate to the reactor. During cold shutdown conditions, reactor water level is maintained with CRD and the Reactor Water Cleanup System (RWCU). These systems are also used to raise and maintain reactor test pressure for the reactor coolant system leakage testing. During cold shutdown conditions, operators closely monitor reactor water level, pressure, and temperature. With the low CRD flowrate, the operators should have sufficient time to react to unanticipated level changes and regain control of reactor pressure, should any abnormalities occur.

The Standby Liquid Control (SLC) System is a high pressure system used to shut down the reactor if the control rods fail to insert. The SLC system has no automatic start function so a spurious start is unlikely. The SLC system must be manually initiated by the use of a keylock switch for each pump.

During cold shutdown conditions, the condensate booster pumps of the Condensate system are shutdown. It would require direct operator action to start a main Condensate Booster system pump and inject into the reactor pressure vessel. The Condensate/Condensate Booster systems are used as an alternate method for filling the RPV and as the primary method for initially pressuring the RPV for pressure testing.

These actions are taken in accordance with procedural guidance that includes verification that RPV coolant and metal temperatures will support filling and pressurizing the RPV with the Condensate/Condensate Booster Pump systems without exceeding the Technical Specification P/T limits.

Low pressure coolant sources include the Emergency Core Cooling Systems (ECCS) (i.e., Low Pressure Core Spray (LPCS) and Low Pressure Coolant Injection (LPCI) systems), and the Condensate system. The shutoff heads of the ECCS pumps and condensate pumps are sufficiently low to preclude a low temperature overpressurization event that would exceed the P/T curve limits and an inadvertent low pressure ECCS injection.

In addition to the procedural barriers, licensed operators are provided specific training on the P/T curves and requirements of the Technical Specifications. Simulator sessions are conducted which include plant heat-up and cool-down. Additionally, in response to industry operating experience, the operator training program is routinely evaluated and revised, as necessary, to reduce the possibility of events such as a low temperature overpressurization event.

Based on the above, procedural and administrative controls, as reinforced in operator training, are in place to effectively limit a low temperature overpressurization event.

Summary

In summary, EGC has reviewed the methodology used in Reference 1, and considering CPS plant specific materials properties, fluence, operational practices, and the provisions of Reference 2, the criteria established in Generic Letter 98-05 (i.e., Reference 3) are satisfied.

Therefore, permanent relief is requested from the examination requirements of 10 CFR 50.55a for RPV circumferential shell welds since the proposed alternative provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

Permanent relief is requested for the remainder of the existing operating license for CPS.

7. Precedents

The NRC has previously approved similar relief for several nuclear power plants, including Dresden Nuclear Power Station, Units 2 and 3 (References 4 and 5), Susquehanna Steam Electric Station, Units 1 and 2 (References 6 and 7), and Quad Cities Nuclear Power Station, Units 1 and 2 (References 9 and 10).

8. References

1. BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 28, 1995
2. Letter from G. C. Lainas (U. S. Nuclear Regulatory Commission) to C. Terry (BWRVIP), "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," dated July 28, 1998
3. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998
4. Letter from J. M. Heffley (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Relief Request for Alterative Weld Examination of Circumferential Reactor Pressure Vessel Shell Welds," dated July 26, 1999
5. Letter from A. J. Mendiola (U. S. Nuclear Regulatory Commission) to O. D. Kingsley (Commonwealth Edison Company), "Dresden - Authorization for Proposed Alterative Reactor Pressure Vessel Circumferential Weld Examinations (TAC Nos. MA6228 and MA6229)," dated February 25, 2000
6. Letter from R. G. Byram (PPL Susquehanna, LLC) to U. S. Nuclear Regulatory Commission, "Request for Alternative to 1 OCFR50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds for PPL Susquehanna LLC Units 1 and 2 PLA-5251," dated November 7, 2000
7. Letter from M. Gamberoni (U. S. Nuclear Regulatory Commission) to R. G. Byram (PPL Susquehanna, LLC), "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)," dated February 28, 2001
8. Letter from J. R. Strosnider (U. S. Nuclear Regulatory Commission) to C. Terry (BWRVIP Chairman), "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. MA3395)," dated March 7, 2000
9. Letter from P. R. Simpson (Exelon), "Relief Request for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for the Fourth Interval Inservice Inspection Program," dated May 16, 2003
10. Letter from A. J. Mendiola (U. S. Nuclear Regulatory Commission) to C. Crane (Exelon), "Quad Cities Nuclear Power Station, Units 1 and 2 – Authorization For Proposed Alternative Reactor Pressure Vessel Circumferential Shell Weld Examination (TAC Nos. MB8985 and MB8986)," dated April 29, 2004