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10 CFR 50.55a

RS-04-032

February 23, 2004

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Extension of Relief for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Additional License Operating Period

References:

(1) Letter from A. J. Mendiola (U. S. NRC) to O. D. Kingsley, "Dresden – Authorization for Proposed Alternative Reactor Pressure Vessel Circumferential Weld Examinations (TAC Nos. MA6228 and MA6229)," dated February 25, 2000

(2) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Relief Request for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Fourth Interval Inservice Inspection Program," dated May 19, 2003

(3) Letter from J. A. Benjamin (Exelon Generation Company, LLC) to U. S. NRC, "Application for Renewed Operating Licenses," dated January 3, 2003

(4) 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

In Reference 1, the NRC approved an alternative reactor pressure vessel (RPV) weld examination pursuant to the provisions of 10 CFR 50.55a, "Codes and standards," paragraphs (a)(3)(i) and (g)(6)(ii)(A)(5) for Dresden Nuclear Power Station (DNPS) Units 2 and 3. The alternative allows permanent deferral of requirements to perform a volumetric examination of circumferential RPV shell welds for the remaining terms of the DNPS Units 2 and 3 operating

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licenses. The approved alternative requires inspections of essentially 100 percent of all longitudinal welds, and inspections of approximately 2 to 3 percent of the circumferential welds at their points of intersection with the longitudinal welds.

In Reference 2, Exelon Generation Company, LLC (EGC) requested similar relief for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2 for the remaining terms of the respective operating licenses. Although the NRC has not yet approved the Reference 2 request, EGC is anticipating approval by May 15, 2004.

Reference 3 requested NRC approval to extend the operating term of the current operating license for each DNPS and QCNPS unit 20 years beyond the current expiration date. In order to allow continued use of the alternative RPV circumferential examinations once the Reference 3 request is approved, EGC requests NRC approval of the attached relief request.

The attached relief request would allow permanent deferral of requirements to perform a volumetric examination of circumferential RPV shell welds for the additional license operating period requested in Reference 3. As an alternative, inspections of essentially 100 percent of all longitudinal welds, and inspections of approximately 2 to 3 percent of the circumferential welds at their points of intersection with the longitudinal welds, would be required.

In Reference 4, 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. The proposed relief request is consistent with the requirements of Reference 4. 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).

EGC requests approval of the attached relief request by February 23, 2005, following the scheduled approval date of our Reference 3 request.

Should you have any questions related to this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager - Licensing

Attachment:
Reactor Pressure Vessel Circumferential Weld Relief Request

ATTACHMENT
Reactor Pressure Vessel Circumferential Weld Relief Request

ASME Components Affected

Components affected are American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Class 1 pressure retaining reactor pressure vessel (RPV) shell circumferential welds, Examination Category B-A, Item No. B1.11.

Applicable Code Edition and Addenda

The applicable ASME Code, Section XI, for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, is the 1995 Edition through 1996 Addenda.

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 additional license operating period requested in Reference 1 for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, from the requirement of ASME Code Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

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).

Reason for Request

Reference 2 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 2, and the results are documented in Reference 3. 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).

Proposed Alternative and Basis for Use

Proposed Alternative

In accordance with 10 CFR 50.55a(a)(3)(i), and consistent with information contained in Reference 4, EGC proposes the following alternate provisions for the subject weld examinations since the proposed alternative provides an acceptable level of quality and safety.

The failure frequency for RPV shell circumferential welds is sufficiently low to justify their elimination from the Inservice Inspection (ISI) requirement of ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

The ISI 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. These welds are generally only accessible from inside surfaces of the RPV using an automated ultrasonic inspection system, which provides the best possible examination of the RPV shell longitudinal welds. Inspections from the outside surfaces have limited access due to the close proximity of the biological shield to the RPV. Also, the reflective insulation that occupies this space is not designed for removal.

Basis for Use

Reference 2 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 2 are documented in Reference 3. Reference 4 permits BWR licensees to 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 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 30, 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 30, 1998, safety evaluation.

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.

Tables 1 and 2 provide a comparison of the limiting RPV circumferential weld parameters for each DNPS and QCNPS unit to those found in Table 2.6-5 of the NRC final safety evaluation of BWRVIP-05 (i.e., Reference 3) for a Babcock and Wilcox vessel. Although the chemistry composition and chemistry factor for DNPS Unit 3 are higher than the limits of the NRC analysis, the shifts in reference temperature for both units are lower than the shift from the NRC limiting analysis. In addition, the unirradiated reference temperatures for both DNPS units are lower. The combination of unirradiated reference temperature and embrittlement shift yields adjusted reference temperatures considerably lower than the NRC mean analysis values.

The chemistry composition and chemistry factor for QCNPS Unit 1 are less than or equal to the limits of the NRC analysis. While the nickel content for Unit 2 is higher than the value utilized in the NRC analysis, the Unit 2 copper content and the chemistry factor are considerably lower than the values utilized in the NRC analysis. Additionally, the unirradiated reference temperatures for both QCNPS units are lower than the NRC limits. The combination of unirradiated reference temperature and embrittlement shift yields adjusted reference temperatures considerably lower than the NRC mean analysis values.

The end of life (i.e., 54 effective full power year (EFPY)) inside diameter fluences for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, are considerably lower than the NRC estimated 54 EFPY fluence. The 54 EFPY fluence estimates were calculated using the fluence methodology of General Electric Nuclear Energy licensing topical report NEDC-32983P (i.e., Reference 5), which was approved by the NRC in Reference 6, and adheres to the guidance of Regulatory Guide 1.190 (i.e., Reference 7). The end of extended license operating time of 54 EFPY includes the extended power uprate approved by the NRC in References 8 and 9. There are no additional uprates planned for DNPS or QCNPS.

The shifts in reference temperature for all four units are lower than the 54 EFPY shift from the NRC analysis. Therefore, for each unit, the RPV shell weld embrittlement due to fluence is calculated to be less than the NRC's limiting case, and each unit's RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, P(FIE), in the NRC's limiting plant specific analysis (54 EFPY) through the projected additional license operating period. For these reasons, the DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, RPVs are bounded by Reference 3.

Table 1: Effects of Irradiation on RPV Circumferential Weld Properties – DNPS			
Parameter Description	**DNPS Unit 2 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 71249/8504)**	**DNPS Unit 3 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 299L44/8650 (WF-19/WF-25))**	**NRC Limiting Plant Specific Analysis (54 EFPY)**
Copper (weight %)	0.23	0.34	0.31
Nickel (weight %)	0.59	0.68	0.59
Chemistry Factor	168	221	196.7
End of Life Inside Diameter Fluence (10^{19} n/cm^2)	0.042	0.041	0.19
ΔRT_{NDT} (°F)	44	58	109.4
$\Delta RT_{NDT(U)}$ (°F)	10	-5	20
Mean RT_{NDT} (°F)	54	53	129.4

Table 2: Effects of Irradiation on RPV Circumferential Weld Properties – QCNPS			
Parameter Description	**QCNPS Unit 1 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # 406L44/8688)**	**QCNPS Unit 2 Parameters at 54 EFPY (Weld Wire Heat/Flux Lot # S3986/3870) Linde 124**	**NRC Limiting Plant Specific Analysis (54 EFPY)**
Copper (weight %)	0.27	0.05	0.31
Nickel (weight %)	0.59	0.96	0.59
Chemistry Factor	183	68	196.7
End of Life Inside Diameter Fluence (10^{19} n/cm^2)	0.041	0.041	0.19
ΔRT_{NDT} (°F)	48	18	109.4
$\Delta RT_{NDT(U)}$ (°F)	-5	-32	20
Mean RT_{NDT} (°F)	43	-14	129.4

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.

Response

EGC has procedures in place for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, that guide operators in controlling and monitoring reactor pressure during all phases of operation, including cold shutdown. Use of these procedures minimizes the potential for low temperature over-pressurization (LTOP) events, and is reinforced through operator training. A Primary System Leakage test is performed prior to each restart after a refueling outage. The associated station test procedure has sufficient guidance to minimize the likelihood of an LTOP event, and requires a pre-job briefing 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 to defeat system actuations that are not required operable. RPV temperature and pressure are required to be monitored and controlled to within the Technical Specifications pressure and temperature (P/T) limits curve during all portions of the testing. The normal and contingency methods to enact pressure control are specified in the test procedure.

A designated Test Coordinator is responsible for the coordination of the test (i.e., from initiation to conclusion) and maintains cognizance of test status. A controlled rate of pressure increase is administratively limited in the test procedure to approximately 30 pounds per square inch (psi) per minute at DNPS, and not greater than 50 psi per minute at QCNPS. If the rate of pressurization exceeds this limit, a contingency sequence portion of the testing procedures provides directions to reduce the rate of pressure increase by depressurizing through the Reactor Water Cleanup System, securing Control Rod Drive (CRD) pumps, and opening the main steam drain lines.

Other than the CRD system, the other high pressure coolant sources that could inadvertently initiate and result in an LTOP event are the Condensate/Feedwater, the Safe Shutdown Makeup Pump (SSMP) at QCNPS, Reactor Core Isolation Cooling (RCIC) at QCNPS, and High Pressure Coolant Injection (HPCI) Systems.

During a normal RPV fill sequence prior to pressure testing, the Condensate System is used to fill the reactor. This evolution is carefully controlled per the test procedure to minimize the potential for an LTOP. The feedwater pump motors are prevented from starting by the reactor water level high feedwater pump trip signal, which is present due to the high reactor water levels required during pressure testing. The SSMP is a manually operated system that has no automatic initiation signals. Initiation of the SSMP is strictly governed by station procedures. During pressure testing, the reactor is in cold shutdown, and as a result, there is no steam available to drive the turbine driven RCIC or HPCI pumps. In addition, the HPCI and RCIC steam supply and pump discharge valves are closed and their associated motor operator breakers are opened in accordance with the test procedures.

The Standby Liquid Control (SLC) system is also a high pressure water source to the RPV. Similar to the SSMP, there are no automatic initiation signals associated with this system. Operation of the SLC system is strictly governed by station emergency operating procedures, and requires an operator to manually start the system from the main control room via a keylock switch manipulation.

The low pressure coolant sources include the Emergency Core Cooling Systems (ECCS) (i.e., Core Spray and Residual Heat Removal) and the Condensate System. Operation of the ECCS systems is also governed by station emergency operating procedures. Although certain automatic initiation signals are required operable during pressure testing, an ECCS actuation would occur only when reactor conditions warranted RPV injection (for example, during a low water level condition). In addition, the shutoff head of the ECCS pumps is relatively low and the injection valves are interlocked closed at pressures greater than approximately 300 psig. For these reasons, an LTOP event that would exceed the P/T curve limits due to an inadvertent ECCS injection is considered unlikely. As mentioned above, the Condensate System is normally used for RPV fill and is carefully governed by the test procedure.

During cold shutdown when the reactor head is tensioned, an LTOP event is prevented by the normal unit shutdown procedure, which requires the operator to place the RPV head vent valves in an open position when reactor coolant temperatures are below 190°F.

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 operating training program is routinely evaluated and revised, as necessary, to reduce the possibility of events such as an LTOP.

Summary

In summary, EGC has reviewed the methodology used in Reference 2, and considering DNPS and QCNPS plant specific materials properties, fluence, operational practices, and the provisions of Reference 3, the criteria established in Generic Letter 98-05 (i.e., Reference 4) are satisfied. Therefore, permanent relief is requested from the examination requirements of the ASME Code Section XI, Subarticle IWB-2500, Table IWB-2500-1, Examination Category B-A, Item No. B1.11 for RPV circumferential shell welds since the proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i).

Duration of Proposed Alternative

Permanent relief is requested for the additional license operating period requested in Reference 1 for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2. Although Reference 4 permits BWR licensees to request permanent relief for the remaining term of the existing initial operating license, EGC has demonstrated that the criteria specified in Reference 4 will continue to be met for the entire additional operating period requested in Reference 1. Therefore, the requested duration of the proposed alternative is justified.

Precedents

The NRC has previously approved similar relief for several nuclear power plants, including Dresden Nuclear Power Station, Units 2 and 3 (i.e., Docket Numbers 50-237 and 50-249, TAC Nos. MA6228 and MA6229), and Susquehanna Steam Electric Station, Units 1 and 2 (i.e., Docket Numbers 50-387 and 50-388, TAC Nos. MB0484 and MB0485). The relief request for Dresden Nuclear Power Station was submitted to the NRC in Reference 10, and the NRC granted the relief in Reference 11. The relief request for Susquehanna Steam Electric Station was submitted to the NRC in Reference 12, and the NRC granted the relief in Reference 13.

References

1. Letter from J. A. Benjamin (Exelon Generation Company, LLC) to U. S. NRC, "Application for Renewed Operating Licenses," dated January 3, 2003

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 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

4. 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

5. NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated August 2000

6. Letter from S. A. Richards (U. S. NRC) to J. F. Klapproth (GE Nuclear Energy), "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation' (TAC No. MA9891)," dated September 14, 2001

7. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001

8. Letter from L. W. Rossbach (U. S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0844 and MB0845)," dated December 21, 2001

9. Letter from S. N. Bailey (U. S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843)," dated December 21, 2001

10. Letter from J. M. Heffley (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Relief Request for Alternative Weld Examination of Circumferential Reactor Pressure Vessel Shell Welds," dated July 26, 1999

11. Letter from A. J. Mendiola (U. S. Nuclear Regulatory Commission) to O. D. Kingsley (Commonwealth Edison Company), "Dresden – Authorization for Proposed Alternative Reactor Pressure Vessel Circumferential Weld Examinations (TAC Nos. MA6228 and MA6229)," dated February 25, 2000

12. Letter from R. G. Byram (PPL Susquehanna, LLC) to U. S. Nuclear Regulatory Commission, "Request for Alternative to 10CFR50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds for PPL Susquehanna LLC Units 1 and 2 PLA-5251," dated November 7, 2000

13. 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