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June 25, 2003

10CFR50.55a(a)(3)

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Subject: LaSalle Station Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Request to Implement CR-38 Addressing Boiling Water Reactor Shell Weld Inspection Recommendations of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report BWRVIP-05

References:

1. EPRI Report TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995.
2. Letter from NRC to the BWRVIP, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. M93925)", dated July 28, 1998.
3. Letter from S. A. Richards (NRC) to J. F. Klapproth (GE-NE), "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891),'" dated September 14, 2001.
4. Letter from K. R. Jury (EGC) to the NRC, "Request for Amendment to Technical Specifications Section 3.4.11, 'RCS Pressure and Temperature (P/T) Limits,'" dated January 31, 2003.
5. Letter from NRC to the BWRVIP, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. MA3395) dated March 7, 2000.
6. 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.
7. Letter from R.M. Krich (EGC) to the NRC, "Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity", dated July 30, 1998.

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In accordance with 10 CFR 50.55a(a)(3), Exelon Generation Company (EGC), LLC, requests approval of proposed Relief Request CR-38 for use at LaSalle County Station, Unit 1 and Unit 2. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety. Relief Request CR-38 requests permanent relief for the remaining term of the operating licenses for Units 1 and 2 from the following requirements:

1. Volumetric examination of all reactor pressure vessel (RPV) shell circumferential welds in the RPV in accordance with the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Examination Category B-A, Item B1.11.
2. Successive inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2420.
3. Additional examinations for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2430.

The technical basis providing justification for relief from the examination requirements of RPV shell circumferential welds is contained in a report submitted from the BWRVIP to the NRC (i.e., Reference 1). The NRC evaluated this report and responses to Requests for Additional Information, and issued safety evaluations to the BWRVIP (i.e., References 2 and 5). Additionally, NRC Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998 (Reference 6), permits BWR licensees to request permanent relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV shell circumferential welds (i.e., ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Shell Welds).

We request that the proposed relief request be approved by December 1, 2003.

Should you have any questions concerning this letter, please contact Mr. Glen T. Kaegi, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

George P. Barnes

George P Barnes
Site Vice President
LaSalle County Station

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station

10 CFR 50.55a Request Number CR-38
Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

1. ASME Code Components Affected

Code Class:	1
Examination Category:	B-A
Item Number:	B1.11
Description:	Relief from Volumetric Examination of All Pressure Retaining Reactor Pressure Vessel Shell Circumferential Welds
Component Number:	Class 1 pressure retaining reactor pressure vessel (RPV) shell circumferential welds.

2. Applicable Code Edition and Addenda

In accordance with the provisions of 10 CFR 50.55a(a)(3), LaSalle County Station requests permanent relief for the remaining term of the operating licenses for Units 1 and 2 from the following requirements:

a. Volumetric examination of all RPV shell circumferential welds in the Reactor Pressure Vessel in accordance with the requirements of ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Examination Category B-A, Item B1.11.

b. Successive Inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2420.

c. Additional Examinations for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2430.

3. Applicable Code Requirement

ASME Section XI 1989 Edition with no addenda is applicable to the ISI Program for the current interval.

4. Reason for Request

LaSalle requests this relief to reduce the number of circumferential welds requiring inspection as endorsed by Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the Requirements on Reactor Pressure Vessel Circumferential Shell Welds".

5. Proposed Alternative and Basis for Use

I. Alternative Provisions:

In accordance with 10 CFR 50.55a(a)(3)(i), LaSalle County Station will implement the following alternate provisions for the subject weld examinations. Unless stated otherwise, all references to the ASME code are to the 1989 Edition of ASME Section XI.

a. Inservice Inspection Scope

The failure frequency for ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel 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 requirements of ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, "Reactor Pressure Vessel Shell Longitudinal," 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% of the RPV shell circumferential welds. EGC believes that when this examination is performed, an automated ultrasonic inspection system will provide the best possible examination of the RPV shell longitudinal welds. Automatic examinations will be implemented where practical and supplemented by manual examinations to maximize volumetric coverage when necessary. 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. Qualification and examination will be completed in accordance with the 1995 Edition 1996 Addenda of ASME Section XI Appendix VIII as modified by the Performance Demonstration Initiative (PDI) and 10CFR50.55(a), "Codes and Standards."

b. Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," at intersections with longitudinal welds, successive examinations per IWB-2420 "Successive Inspections," 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 (Reference 1),

2. The NDE 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 ASME Section XI, 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, "Reactor Pressure Vessel Shell Longitudinal Welds," all flaws shall be reinspected at successive intervals consistent with ASME Code and regulatory requirements.

c. Additional Examinations of Flaws

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," at the intersection with longitudinal welds, additional requirements per ASME Section XI, 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 (Reference 1) then no additional examinations are required.

2. If the flaw is not characterized as subsurface in accordance with BWRVIP-05 (Reference 1) 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,
   - An evaluation per ASME Section XI, IWB-3600 demonstrating that the vessel is acceptable for continued service.

3. If the flaw meets the criteria of ASME Section XI, IWB-3600 for 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 ASME Section XI, IWB-2430(a). 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 additional examination for flaws shall be in accordance with ASME Section XI, IWB-2430, "Additional Examinations." 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.

II. Basis for Relief

The technical basis providing justification for relief from the examination requirements of RPV shell circumferential welds is contained in a report submitted from the BWRVIP to the NRC (i.e., Reference 1). The NRC evaluated this report and responses to Requests for Additional Information, and issued Safety Evaluations to the BWRVIP (i.e., References 2 & 5.). Additionally, NRC Generic Letter (GL) 98-05 (i.e., Reference 6) permits BWR licensees to request permanent relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV shell circumferential welds (ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Shell Welds). This relief can be granted by demonstrating two criteria: (1) at the expiration of their license, the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the NRC Safety Evaluation (Reference 2), and (2) licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC Safety Evaluation (Reference 2).

Criterion 1, Demonstrate that at the expiration of the license, the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the July 30, 1998 Safety Evaluation.

The NRC evaluation of BWRVIP-05 utilized a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis 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 each RPV to those found in Table 2.6-4 of the NRC Final Safety Evaluation of BWRVIP-05. LaSalle County Station Unit 1 RPV was manufactured by Combustion Engineering (CE) and Unit 2 RPV was manufactured by Chicago Bridge and Iron Company (CB&I). The CE vessel chemistry limits included in the "NRC Limiting Plant Specific Analysis (32 EFPY)" column are those from the NRC Safety Evaluation (Reference 2) as reported in the Combustion Engineering Owner's Group (CEOG) Report, Reference 22 of Reference 2. The basis for the LaSalle Unit 1 and Unit 2 copper and nickel values in Table 1 is the LaSalle Station response to the NRC Request for Additional Information associated with Generic Letter 92-01, "Reactor Vessel Structural Integrity" (Reference 7).

LaSalle County Station requested a change to the Reactor Coolant System Pressure and Temperature (P/T) Limits Technical Specifications of Unit 1 and 2 (Reference 4). The 32 Effective Full Power Year (EFPY) fluence was calculated using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation", which was approved by the NRC in Reference 3. Additionally, as a conservative assumption, the 32 EFPY fluences were calculated at the inside surface of the reactor vessel wall using the current uprated power and twenty-four month fuel management designs throughout the entire operational life of LaSalle County Station. The Reference 4 submittal is currently under review by the NRC, and the fluence results of these calculations were used in this submittal.

The summary of the evaluation of the Unit 1 RPV is shown on Table 1. The copper content for Unit 1 is higher than the value utilized in the NRC analysis, however, the Unit 1 nickel content and chemistry factor are considerably lower than the values utilized in the NRC analysis. The unirradiated reference temperature is lower than that used in the NRC analysis. The calculated 32 EFPY fluence for Unit 1 is considerably lower than the NRC estimated values. Overall, the relatively high copper content on Unit 1 compared with the content used in the NRC analysis is more than compensated for by the lower nickel content, unirradiated temperature and fluence; resulting in a lower calculated mean reference temperature than the NRC mean analysis values.

The summary of the evaluation of the Unit 2 RPV is also shown on Table 1. From the table, the Unit 2 chemistry composition and chemistry factor are lower than the values used in the NRC analysis. The unirradiated reference temperature is higher than that used in the NRC analysis for Unit 2. The calculated 32 EFPY fluence for Unit 2 is considerably lower than the NRC estimated values. For Unit 2, the relatively higher unirradiated reference temperature compared with the NRC limit is more than compensated for by lower copper and nickel content and fluence, resulting in a lower calculated mean reference temperature than the NRC mean analysis values.

The RPV shell circumferential weld RT_{NDT} due to fluence is calculated to be less for each Unit than the NRC's limiting case and therefore each Unit's RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, P(FIE), in Table 2.6-4 of the NRC Safety Evaluation (Reference 2) through the initial end of license.

Table 1, Effects of Irradiation on RPV Circumferential Weld Properties
LaSalle Station Units 1 and 2

Parameter Description	LCS Unit 1 Parameters at 32 EFPY	LCS Unit 2 Parameters at 32 EFPY	NRC Limiting Plant Specific Analysis (32 EFPY)	
	CE RPV	CB&I RPV	CE RPV	CB&I RPV
Copper, wt. %	0.205	0.04	0.183	0.10
Nickel, wt %	0.105	0.94	0.704	0.99
Chemistry Factor	98	54	172.2	134.9*
End of Life Inside Diameter Fluence, X 10^{19} n/cm^2	0.102	0.109	0.20	0.51
ΔRT_{NDT}, °F	41.2	23.5	98.1	109.5
$RT_{NDT(U)}$, °F	-50	-34	0	-65
Mean RT_{NDT}, °F	-8.8	-10.5	98.1	44.5

* Revised value from the Reference 5 letter.

Criterion 2, Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the July 30, 1998 Safety Evaluation.

EGC has procedures in place for LaSalle County Station 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 will prevent a Low Temperature Over-Pressurization (LTOP) event, and are reinforced through operator training. Operating Procedures have sufficient guidance to prevent a LTOP event. A reactor vessel pressure test is performed prior to each restart after a refueling outage. This procedure requires an Operations briefing prior to test commencement with all involved personnel. Vessel temperature and pressure are required to be monitored and controlled to within the Technical Specification pressure-temperature (P-T) curve during all portions of testing. The normal and contingency methods to enact pressure control are specified.

A Senior Reactor Operator, who is designated as a Test Coordinator during cold pressure testing, is responsible for the coordination of the test from initiation to conclusion and maintains cognizance of test status. A controlled rate of pressure increase is administratively limited in the test procedure to no greater than 50 psi per minute. If the rate of pressurization exceeds this limit, contingencies are included to reduce the rate of pressure increase by depressurizing through the Reactor Water Clean Up system and by securing the Control Rod Drive (CRD) pump.

Other than the CRD system, the high pressure coolant sources that could inadvertently initiate and result in a LTOP event are the Feedwater, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Spray (HPCS) systems. During a normal RPV fill sequence prior to pressure testing, the condensate system is used to fill the reactor. The Motor Driven Reactor Feedpump (MDRFP) is prevented from starting by the high level Feedwater pump trip signal, which is present due to the high reactor water levels

required during pressure testing. During pressure testing, 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 Feedpumps (TDRFPs). The HPCS pump control switch is placed in pull-to-lock to ensure an inadvertent initiation will not occur.

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

During cold shutdown when the reactor head is tensioned, a LTOP event is prevented by the operating shutdown procedure, which requires the operator to place the RPV head vent valves in an open position when reactor coolant temperatures are below 212°F.

In addition to the procedural barriers, licensed operators are provided specific training on the P-T curves and the associated requirements of the Technical Specifications. Simulator sessions are conducted which focus on plant heat-up and cool-down and equipment surveillance where adherence to these examinations is required. Additionally, in response to industry operating experience and events, the Operations Training instructors and staff routinely evaluate and develop operating training programs to reduce the possibility of events such as LTOP.

In summary, Exelon has reviewed the methodology used in BWRVIP-05 (Reference 1), and considering LaSalle County Station plant specific materials properties, fluence and operational practices, and the provisions of the NRC Safety Evaluation Report (Reference 2), EGC believes the criteria established in Generic Letter 98-05 are satisfied. Therefore, permanent relief is requested from the examination requirements of 10 CFR 50.55a for reactor pressure vessel 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 remaining term of the operating licenses of Units 1 and 2.