January 28, 2004

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 - RELIEF REQUEST CR-38
RE: SHELL WELD INSPECTION (TAC NOS. MB9755 AND MB9756)

Dear Mr. Skolds:

By letter dated June 25, 2003, Exelon Generation Company, LLC, submitted a request for permanent relief from volumetric examination of all pressure vessel circumferential weld inspection requirements of American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The Relief Request CR-38 by LaSalle County Station, Units 1 and 2, proposes as an alternative to ASME Section XI to use the methodology contained in Electric Power Research Institute (EPRI) report TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 1995, which was approved by the Nuclear Regulatory Commission (NRC) on July 30, 1998.

The NRC staff has evaluated Relief Request CR-38, and finds that the proposed alternative may be authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that it provides an acceptable level of quality and safety. Permanent relief is authorized for the remaining term of the initial operating licenses for LaSalle County Station, Units 1 and 2, consistent with the commitments specified by the licensee and as discussed in the enclosed safety evaluation.

Sincerely,

***/RA by DPickett for/***

Anthony J. Mendiola, Chief, Section 2
Project Directorate, III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos.: 50-373 and 50-374

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INSERVICE INSPECTION REQUIREMENTS

RELIEF REQUEST CR-38

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

## 1.0 INTRODUCTION

By application dated June 25, 2003 (Ref. 1), Exelon Generation Company, LLC, submitted a request for permanent relief from volumetric examination of all pressure vessel circumferential weld inspection requirements of American Society of Mechanical Engineers (ASME) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Relief Request CR-38 for LaSalle County Station, Units 1 and 2 (LSCS), proposes as an alternative to ASME Section XI to use the methodology contained in Electric Power Research Institute (EPRI) technical report TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 1995 (Ref. 2), which was approved by the Nuclear Regulatory Commission (NRC) on July 30, 1998 (Ref. 3).

The proposed alternative would reduce the number of circumferential welds requiring inspection as endorsed by Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the Requirements on Reactor Pressure Vessel Circumferential Shell Welds" (Ref. 4). Specifically, the proposed alternative would apply the methodology used in BWRVIP-05 to satisfy the provisions of GL 98-05.

## 2.0 REGULATORY EVALUATION

The staff finds that the licensee in sections 2 and 3 of its submittal identified the applicable regulatory requirements. The regulatory requirements for which the staff based its acceptance are provided below.

Pursuant to Section 50.55a(g)(4) of Title 10 of the *Code of Federal Regulations* (10 CFR), components classified as ASME Code Class 1, 2 and 3 must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the ASME Code to the extent practical within the limitations of design, geometry and materials of construction of the components. The regulations require that all inservice examinations and system pressure tests conducted during the first 10-year interval and subsequent intervals on ASME Code Class 1, 2, and 3 components must comply with the requirements in the latest edition and addenda of Section XI incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 10-year interval. The applicable edition of Section XI for LaSalle, Units 1 and 2, during the current 10-year ISI interval is the 1989 Edition, with no addenda.

10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used when authorized by the NRC if: "(i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

GL 98-05 allows BWR licensees to request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, “Circumferential Shell Welds”) by demonstrating that:

a. 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 (SE) (Ref 4.), and
b. 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, SE.

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

## 3.0 TECHNICAL EVALUATION

The staff has reviewed the licensee’s proposed alternative and basis for use in support of its proposed relief request which are described in Section 5 of the licensee’s submittal. The detailed evaluation below will support the conclusion that the licensee-proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i).

### 3.1 Code Requirements:

ASME Code, Section XI, 1989 Edition, Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item B1.10, Examination Category B-A, “Pressure Retaining Welds in Reactor Vessel,” requires examinations of essentially 100 percent of the RPV shell welds. ASME Code category B1.10 covers requirements for examinations of RPV circumferential shell welds (Examination Item B1.11) and longitudinal shell welds (Examination Item B1.12).

### 3.2 Specific Relief Requested:

In accordance with the provisions of 10 CFR 50.55a(a)(3) and consistent with the staff’s SE dated July 30, 1998, for the BWRVIP-05 report, the licensee is requesting permanent relief for the remaining term of the operating licenses for LSCS from the following requirements:

1. Volumetric examination of all RPV shell circumferential welds in the RPV 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.

2. Successive Inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2420.
3. Additional Examination for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 1989 Edition, Paragraph IWB-2430.

3.3 Basis for relief:

Pursuant to 10 CFR 55.55(a)(3)(i), and consistent with information contained in GL 98-05, the licensee is proposing an alternative to ASME Code, Section XI, requirements to examine essentially 100 percent of accessible Category B-A circumferential welds and is proposing permanent relief (for the remaining portion of the initial license period) from these examinations.

Consistent with GL 98-05, the licensee provided the following information:

1. At expiration of license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's SE dated July 30, 1998. The NRC evaluation of BWRVIP-05 utilized a probabilistic fracture mechanics analysis to estimate the limiting plant-specific case of BWR RPVs failure probabilities. Although BWRVIP-05 provides the technical basis supporting the relief request, the information provided in Table 1 (next page) shows that LSCS's vessels are bounded by the NRC analysis.

   The staff considers that when the mean (irradiated) reference temperature ($RT_{ndt}$) value for a RPV shell weld is less than the mean $RT_{ndt}$ value for its correspond limiting plant reference case study (as specified in table above), the shell weld is considered to have less embrittlement than the corresponding weld in the case study, and therefore to have a conditional probability of failure less than or equal to that calculated for the reference case study.

2. LSCS has 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 SE.

3.4 Proposed Alternative:

Consistent with the staff's SE dated July 30, 1998, for the BWRVIP-05 report and in accordance with 10 CFR 50.55a(a)(3)(i), LSCS will implement the following alternate provisions for the subject weld examinations:

a. Inservice Inspection Scope

The failure frequency for ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, is sufficiently low to justify their elimination from the ISI requirement of 10 CFR 50.55a(g) based on the SE dated July 30, 1998.

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

| **Parameter Description** | **Unit 1 Parameters at 32 EFPY** | **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, $x10^{19}$ n/cm$^2$ | 0.102 | 0.109 | 0.20 | 0.51 |
| Initial (unirradiated) Reference Temperature $RT_{NDT(U)}$, °F | -50 | -34 | 0 | -65 |
| Increase in Reference Temperature $\Delta RT_{NDT}$, °F | 41.2 | 23.5 | 98.1 | 109.5 |
| Mean (irradiated) Reference Temperature $RT_{NDT(u)} + \Delta RT_{NDT}$ | -8.8 | -10.5 | 98.1 | 44.5 |

* Revised value from the Ref. 5 letter.

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 and shall include inspection of the circumferential welds only at the intersection of these welds with the longitudinal welds, or approximately 2 percent to 3 percent of the RPV shell circumferential 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.

b. Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, 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;
2. The non-destructive examination (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, 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, 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 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 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.

3.5 Evaluation of Relief Request CR-38:

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 staff's SE dated July 30, 1998. The staff has reviewed the licensee's regulatory and technical analyses described in their submittal to determine if the licensee's proposed alternative provides an acceptable level of quality and safety.

3.5.1 Criterion 1 of GL 98-05:

The staff's SE 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 Combustion Engineering (CE) & Chicago Bridge and Iron Works (CB&I), using the highest mean $RT_{NDT}$ to determine the limiting case. Since the LSCS RPVs were fabricated by CE and CB&I, respectively, the licensee compared the mean irradiated $RT_{NDT}$ of both vessels to that for the limiting CE and CB&I cases described in Table 2.6-4 of the staff's SE for the BWRVIP-05 report. (A copy of Table 2.6-4 is included with this safety evaluation as an appendix.) As indicated in the licensee's evaluation, the mean $RT_{NDT}$ for the LSCS vessels are lower than that for the limiting CE and CB&I cases; therefore, the licensee concluded that the conditional failure probability for the LSCS vessels circumferential welds is bounded by the conditional failure probabilities in the staff's SE through the end of the current license period.

The staff's SE provides a limiting conditional failure probability of $6.34 \times 10^{-5}$ per-reactor-year for a limiting plant-specific mean $RT_{NDT}$ of 98.1 °F for CE-fabricated RPVs and a limiting conditional failure probability of $2.00 \times 10^{-7}$ per-reactor-year for a limiting plant-specific mean $RT_{NDT}$ of 44.5 °F for CB&I-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 LSCS is projected to be -8.8 °F for Unit 1 and -10.5 °F for Unit 2 at the end of the current license. In this evaluation, the chemistry factor, $\Delta RT_{NDT}$, and mean $RT_{NDT}$ were calculated consistent with the guidelines of Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988 (Ref. 6). The calculated values of mean $RT_{NDT}$ for the circumferential welds at LaSalle, Units 1 and 2 are significantly lower than that for the limiting plant-specific case for CE and CB&I fabricated RPVs, respectively, indicating that the conditional failure probability of the LaSalle circumferential welds is much less than $6.34 \times 10^{-5}$ per-reactor-year for Unit 1 and much less than $2.00 \times 10^{-7}$ per-reactor-year for Unit 2. Therefore, each unit's RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, P(F|E), in Table 2.6-4 of the staff's SE for the BWRVIP-05 report through the initial end of license.

The 32 Effective Full Power Year (EFPY) fluences were 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 Ref. 7. 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 LSCS, Units 1 and 2.

The staff's plant specific analysis for 32 EFPY fluence value resulted in $2.0 \times 10^{18}$ n/cm$^2$ and $5.1 \times 10^{18}$ n/cm$^2$ for LSCS, Units 1 and 2, respectively. The staff recently evaluated a proposed extension of the pressure-temperature (P-T) limit curves for both LSCS units (Ref. 8). From this evaluation, the 32 EFPY fluence values accepted by the staff were $1.02 \times 10^{18}$ n/cm$^2$ and $1.09 \times 10^{18}$ n/cm$^2$, respectively. Acceptability was based on the fact that the methodology followed the guidance in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (Ref. 9). The calculated values account for the current uprated power level and a load factor adjusted to the 24 month fuel cycle. Comparison of the accepted values for 32 EFPYs and the limiting values indicate that the accepted values are significantly lower than the limiting values, and therefore, are acceptable.

3.5.2 Criterion 2 of GL 98-05:

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.

The licensee has procedures in place for LSCS that guide operators in controlling and monitoring reactor pressure during all phases of operation. Use of the guidance provided in the operating procedures will prevent a Low Temperature Over-Pressurization (LTOP) event. Also, according to the licensee, these procedures are reinforced through operator training.

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 P-T limit curve during all portions of testing. 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 in the procedures 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 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. Also, 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. The HPCS pump control switch is placed in pull-to-lock to ensure an inadvertent initiation will not occur.

Low pressure coolant sources that could inadvertently actuate 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 produce an 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, an LTOP event is prevented by the operating shutdown procedure. This procedure requires the operator to place the RPV head vent valves in an open position when reactor coolant temperature is 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 LTOPs.

The staff has determined that the licensee has performed an acceptable review of the methodology used in BWRVIP-05, as well as the provisions of the NRC Safety Evaluation Report, considering LaSalle plant specific materials properties and end-of-life fluences. Also, the staff reviewed the licensee operational practices, which provide sufficient assurance that it is unlikely that a non-design basis cold over-pressure transient will occur. After evaluation of the licensee's submittal, the staff believes that LSCS satisfied the criteria established in GL 98-05. Therefore, based on the above, the staff concludes that permanent relief requested from the examination requirements of 10 CFR 50.55a(g) for reactor pressure vessel circumferential shell welds, is acceptable under 10 CFR 50.55a(a)(3)(i), since the proposed alternative provides an acceptable level of quality and safety.

All other requirements of the AMSE Code, Section XI, for which relief has not been specifically requested remain applicable, including third party review by the authorized nuclear inservice inspector.

## 4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that the licensee-proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). Permanent relief is authorized for the remaining term of the initial operating licenses for LaSalle County Station, Units 1 and 2, consistent with the commitments specified by the licensee and discussed in this safety evaluation.

## 5.0 REFERENCES

1. Letter from G.P. Barnes (Exelon) to NRC, “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," dated June 25, 2003.

2. Technical report TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Electric Power Research Institute, dated September 1995.

3. Letter from NRC to BWRVIP, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05," dated July 30, 1998.

4. Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the Requirements on Reactor Pressure Vessel Circumferential Shell Welds," U.S. Nuclear Regulatory Commission, dated November 10, 1998.

5. Letter from NRC to BWRVIP, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05," dated March 7, 2000.

6. Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, U.S. Nuclear Regulatory Commission, dated May 1988.

7. 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," dated September 14, 2001.

8. Letter from K.R. Jury (Exelon) to NRC, "Request for Amendment to Technical Specifications Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits,"" dated January 31, 2003.

9. Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, dated March 2001.

Principal Contributors: J. Arroyo, NRR/DSSA/EMCB-A
L. Lois, NRR/DSSA/SRXB-B

Date: January 28, 2004

**APPENDIX**
**FOR INFORMATION USE ONLY**

Table 2.6-4 of the staff's safety evaluation for the BWRVIP-05 report (Ref. 3)
Summary of results of NRC staff and BWRVIP limiting plant-specific analyses 32 EFPY

| FLAW ORIENT. | GROUP | Cu | Ni | CF | FLUENCE ($10^{19}$ n/cm$^2$) | $\Delta RT_{NDT}$ (°F) | $RT_{NDT(U)}$ (°F) | MEAN $RT_{NDT}$ * (°F) | P(F\|E) | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | STAFF | BWRVIP |
| AXIAL | CE (VIP)[a] | 0.26 | 1.20 | 276.0 | 0.15 | 138.8 | -20 | 118.8 | 2.94 E-1 | 1.37 E-2 |
| | CE (CEOG)[b] | 0.219 | 0.996 | 231.1 | 0.20 | 131.6 | 0 | 131.6 | 4.37 E-1 | ----- |
| | CB&I | 0.10 | 1.08 | 135.0 | 0.69 | 121.0 | -30 | 91.0 | 1.42 E-1 | 1.55 E-2 |
| | B&W | 0.25 | 0.35 | 142.5 | 0.125 | 66.0 | 10 | 76.0 | 5.98 E-2 | 8.12 E-3 |
| CIRC. | CE (VIP)[a] | 0.13 | 0.71 | 151.7 | 0.20 | 86.4 | 0 | 86.4 | 2.81 E-5 | NF ($10^6$) [c] |
| | CE (CEOG)[b] | 0.183 | 0.704 | 172.2 | 0.20 | 98.1 | 0 | 98.1 | 6.34 E-5 | ----- |
| | CB&I | 0.10 | 0.99 | 134.9 | 0.51 | 109.5 | -65 | 44.5 | 2 E-7 | 1 E-6 |
| | B&W | 0.31 | 0.59 | 196.7 | 0.095 | 79.8 | 20 | 99.8 | 8.17 E-5 | 1 E-6 |

[a] Chemistry information reported in BWRVIP-05.
[b] Chemistry information reported in CEOG report.
[c] No failures in the indicated number of vessel simulations.
* Mean $RT_{NDT}$ was determined using the peak neutron fluence for the limiting weld.