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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 No. 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: Additional Information for the Review of the License Renewal Applications for Quad Cities Nuclear Power Station, Units 1 and 2 and Dresden Nuclear Power Station, Units 2 and 3

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) Letter from Tae Kim (USNRC) to John Skolds (Exelon Generation Company, LLC), "Request for Additional Information for the Review of the Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Unit 1 and 2, License Renewal Application," dated August 7, 2003

(3) Letter from Tae Kim (USNRC) to John Skolds (Exelon Generation Company, LLC), "Supplemental Request for Additional Information for the Review of the Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Unit 1 and 2, License Renewal Application," dated September 9, 2003

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Exelon Generation Company, LLC (EGC) is submitting the additional information requested in References 2 and 3. This additional information provides further discussion of Section 3.1, "Aging

Management of Reactor Vessel, Internals, and Reactor Coolant System," Section 4.2, "Neutron Embrittlement of the Reactor Vessel and Internals," Section 4.7, "Other Plant-Specific TLAAs," Appendix B, "Aging Management Programs" and Section B.1.8, "BWR [boiling water reactor] penetrations" to support the NRC review of Reference 1.

Should you have any questions, please contact Al Fulvio at 610-765-5936.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

October 3, 2003
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing

Attachment: Response to Request for Additional Information

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

Attachment

Response to Request for Additional Information

DRESDEN AND QUAD CITIES
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

RAI 3.1-1

LRA Table 3.1-1 (Ref. Nos. 3.1.2.37 and 3.1.2.59) identifies cracking as an applicable aging effect for vessel head enclosure cladded with austenitic stainless steel, but NUREG-1801 does not. Provide industry-wide and plant-specific operating experience with the cladded vessel head enclosures that have had cracking. Identify locations where cracking has occurred (cladding, weld metal, base metal). Describe the methodology for detecting cracking and monitoring the crack growth. If cracking will not be repaired prior to the end of the current license, provide an analysis or inspection program that will monitor the crack. Evaluate this to 10 CFR Part 54.3, TLAA criteria. Provide basis for concluding ISI program will detect cracks.

Response:

In 1990, Quad Cites Unit 2 visually detected defects (stain patches) at various points on the RPV head cladding. Dye penetrant and UT examinations were performed to determine the extent of the defects. The defects (cracks) were a maximum depth of approximately 6 mm in the base material. The cracking was attributed to IGSCC and possibly hot cracking. Subsequent examinations in 1990, 1992, and 1995, using Ultrasonic Through-Wall Sizing, VT-1 and VT-3 methods, have indicated no change (no evidence of growth, increased severity or decrease in component integrity).

In 1992 Vermont Yankee observed rust patches in the RPV head. This inspection was performed to address the Quad Cites Unit 2 operating experience. The indications were located primarily in the area of the flange, which had been clad by manual welding. There was no evidence of cracking in the base material. The indications were fine, branched cracks in the cladding, which is consistent with IGSCC.

Dresden and Quad Cities will continue to monitor the RPV Head cladding using the VT methods described in ASME Section XI, IWB-2500-1, Item B13.10. Once cracking occurs in the cladding, the ferritic material under the cladding becomes exposed to the reactor water and steam environment and begins to oxidize or rust. The rust seeps back through the cracked surface providing a readily detectable stain. The visual examination required by ASME Section XI was the method used to detect the evidence of cracking (stain patches) at both Quad Cities and at Vermont Yankee.

The cracking was evaluated as a potential TLAA in accordance with 10CFR Part 54.3. The evaluation concluded this was not a TLAA because the analysis did not involve time-limited assumptions defined by the current operating term at Dresden and the analysis was not contained or incorporated by reference in the current license basis at Quad Cities.

RAI 3.1-2

The applicant identifies cumulative fatigue damage as an applicable aging effect for nozzles and their safe ends, vessel penetrations, support skirts and attachment welds, top head flanges, vessel flanges, vessel shells (including upper shell, intermediate nozzle shell, intermediate beltline shell, and lower shell), and vessel bottom heads. Please confirm whether this identification of cumulative fatigue damage as an aging effect applies to all four units (Dresden Units 2 and 3, and Quad Cities Units 1 and 2). Otherwise, provide technical explanation. The staff raises this question because Table 2.3.1-1 of Aging Management Review Aid provided by the applicant identifies cumulative fatigue damage as an aging effect for support skirts exposed to ambient temperature air and not for the ones exposed to containment nitrogen. The applicant also needs to identify the containment environment in each unit.

Response:

The aging effect of cumulative fatigue damage does apply to all four units (Dresden Units 2 and 3, and Quad Cities Units 1 and 2).

At all four units, the primary containment (drywell and suppression pool) atmosphere is made inert with nitrogen to render the primary containment atmosphere non-flammable by maintaining the oxygen content below 4% by volume during normal operation. The drywell has an average temperature of 135°F during normal operations. The relative humidity in the drywell ranges between 20% and 90%.

The Aging Management Reference 3.1.1.1 for support skirt and attachment welds in LRA Table 3.1-1 reflects an environment of "ambient temperature air" to maintain consistency with the environment of these components as listed in NUREG-1801, Item IV.A1.7-a.

RAI 3.1-3

The applicant does not identify cumulative fatigue damage as an applicable aging effect for stabilizer brackets, the external attachment weld between reactor pressure vessel and refueling bellows, and reactor vessel closure studs, but BWRVIP-74 does identify. In addition, the applicant does not identify cumulative fatigue damage as an applicable aging effect for closure bolting, but NUREG 1801 (Item C1.2-f, Chapter IV.C1) does. Explain why cumulative fatigue damage is not identified as an applicable aging effect for stabilizer brackets, the external attachment weld between reactor pressure vessel and refueling bellows, reactor vessel closure studs, and closure bolting. If cumulative fatigue damage is identified as an aging effect for these components, provide an appropriate program for managing this effect.

Response:

Item C1.2-f, Chapter IV.C1, of NUREG-1801 is the closure bolting for the recirculation pump. The aging effect of cumulative fatigue for the recirculation pumps' closure bolting is shown in the LRA Table 3.1-1, Aging Management Reference 3.1.1.1, which links to the Closure Bolting line of LRA Table 2.3.1-5 (Component Groups Requiring Aging Management Review – Recirculation System).

LRA Section 4.3.1, Reactor Fatigue Analysis, identifies the reactor vessel closure studs as components that may experience cumulative fatigue damage. The reactor vessel closure studs are included in the fatigue monitoring program that is described in Section 4.3.1 of the LRA. LRA Table 2.3.1-1 (Component Groups Requiring Aging Management Review –Reactor Vessel) should have included Aging Management Reference 3.1.1.1 in the Top Head Enclosure (Closure Studs and Nuts) line.

For Dresden and Quad Cities there are no current licensing basis time limited aging analyses (TLAAs) that evaluate cumulative fatigue of the RPV stabilizer brackets or of the external attachment weld between reactor pressure vessel and refueling bellows.

The RPV stabilizer brackets are shown as Support Members in Table 2.4-15 (Component Groups Requiring Aging Management Review –Supports) and are linked to LRA Table 3.5.1, Aging Management Reference 3.5.1.31. These RPV stabilizer brackets are included in the ISI programs at both Dresden and Quad Cities.

The refueling bellows attached to the reactor pressure vessel prevents leakage from the flooded reactor cavity into the drywell during refueling operations. However, the function of preventing leakage into the drywell during refueling operations is not a safety related function and failure of the vessel-to-bellows weld cannot cause failure of a safety related function. The refueling bellows are not within the scope of license renewal. Consequently, the external attachment weld between the reactor pressure vessel and the refueling bellows is not within the scope of license renewal.

LRA Table 2.3.1-1 should have read as follows:

Component	Component Intended Function	Aging Management Ref
Top Head Enclosure (Closure Studs and Nuts)	**Pressure Boundary**	3.1.1.1, 3.1.1.8

RAI 3.1-4

The applicant identifies loss of fracture toughness as an applicable aging effect for reactor pressure vessel flange, intermediate beltline shell, beltline welds, intermediate nozzle shell, lower shell, and upper shell. Identify which components are expected to have neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV) by the end of the extended period of operation. Provide adequate information regarding the AMP to manage loss of fracture toughness for these materials.

Response:

The following components are expected to have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV) by the end of the extended period of operation:

- Lower Shell

- Intermediate Beltline Shell
- Axial Welds in Lower Shell and Intermediate Beltline Shell
- Girth Weld between Lower Shell and Intermediate Beltline Shell

The AMP for the components subject to loss of fracture toughness is B.1.22, "Reactor Vessel Surveillance".

RAI 3.1-5

The LRA identifies no aging effect for the external surface of carbon steel reactor vessel components, vessel head vent system, and nuclear boiler instrumentation system exposed to containment nitrogen environment. The BWR containment environment typically has high humidity. The carbon steel components exposed to this environment may experience loss of material due to corrosion. Explain why loss of material is not considered as an aging effect for these components, or provide a program for managing such effect.

Response:

The drywell is made inert with nitrogen to render the primary containment atmosphere non-flammable by maintaining the oxygen content below 4% by volume during normal operation. The drywell has an average temperature of 135°F during normal operations. The relative humidity in the drywell ranges between 20% and 90%.

According to EPRI 1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3, loss of material due to corrosion is not considered a credible aging effect for carbon steel components in a containment nitrogen environment because of the negligible amounts of free oxygen (< 4%). Both oxygen and moisture must be present for general corrosion to occur because oxygen alone or water free of dissolved oxygen (high humidity in a nitrogen atmosphere) does not corrode carbon steel to any practical extent.

RAI 3.1-6

The LRA identifies no aging effect for the carbon steel drain line penetrations exposed to reactor coolant water up to 288 °C (550 °F). Such drain line is likely to experience loss of material due to corrosion. This assessment is consistent with Item D2.1-a, Chapter V.D2 of NUREG-1801. Explain why loss of material due to corrosion is not considered as an aging effect for these components, or provide a program for managing such effect.

Response:

The drain line penetration is an unclad hole drilled into the reactor vessel bottom head with an unclad carbon steel nozzle welded to the outside of the vessel bottom head. Aging Management Reference 3.1.2.58 should have shown loss of material/general, pitting, and crevice corrosion as an applicable aging effect.

LRA Table 3.1-2 Aging Management Reference 3.1.2.58 should have read as follows:

Ref No	Component Group	Material	Environment	Aging Effect/Mechanism	Aging Management Program	Discussion
3.1.2.58	Penetrations Drain Line Nozzles	Carbon Steel	Up to 288°C (550°F) reactor coolant water	Loss of material/general, pitting, and crevice corrosion	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1), for Class 1 components; and Water Chemistry (B.1.2)	NUREG-1801 does not address carbon steel penetrations in a reactor coolant water environment.

RAI 3.1-7

(a) D/QCNPS has used extended power uprates to increase the power output of each of the four units by about 17 to 18%. Such increase in power may increase the fluence on vessel internals and reactor vessel wall. Explain how this increase in power has been accounted for in performing aging management review of vessel internals and reactor vessel shell. The "Final License Renewal SER for BWRVIP-26," dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm^2 (E > 1 MeV). Identify the vessel internals whose fluence at the end of extended period of operation with power uprate conditions may exceed the threshold level and become susceptible to cracking due to IASCC. What AMP will be utilized to manage IASCC of the components that exceed the threshold?

(b) The reactor vessel internals that may receive neutron fluence greater than the threshold fluence for IASCC [5×10^{20} n/cm^2 (E > 1 MeV)] by the end of extended period of operation are susceptible to cracking due to IASCC. According to the "Final License Renewal SER for BWRVIP-26," dated December 7, 2000, the accumulated neutron fluence is a TLAA issue for these vessel internals. The SER for BWRVIP-26 further states that the applicant must identify and evaluate this TLAA issue. Provide identification and evaluation of the accumulated neutron fluence received by the D/QNPS vessel internals at the end of the extended license period as a TLAA issue.

Response:

(a) The fluence calculations prepared specifically for the Dresden and Quad Cities license renewal application included the effects of extended power uprate. The top guide, shroud, and the in-core instrumentation guide tubes and dry tubes may exceed the threshold fluence value of 5×10^{20} n/cm^2 (E > 1 MeV) by the end of the period of extended operation. As such, these components will require aging management. The AMPs used to manage the IASCC aging effect are B.1.2, "Water Chemistry," and B.1.9, "BWR Vessel Internals."

(b) As stated above, fluence calculations were prepared for the reactor vessel and internals, including the effects of extended power uprate. Three components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; and (3) In-core Instrumentation Dry Tubes and Guide Tubes. As such, these components will require aging management as discussed above. However, contrary to the direction contained in the SER for BWRVIP-26, this technical issue does not qualify as a Time Limited Aging Analysis (TLAA). Specifically, the analysis is not contained or incorporated by reference in the current licensing basis for either site. As such, it does not satisfy Criterion (6) of10 CFR 54.3, Definitions, Time Limited Aging Analyses. Dresden and Quad Cities Stations will implement the BWRVIP recommendations, and manage the effects of aging of IASCC through aging management programs B.1.2 (Water Chemistry), and B.1.9 (BWR Vessel Internals).

RAI 3.1-8

The applicant credits ASME Section XI inservice inspection program for managing cracking in the welded access hole covers due to SCC. This program requires visual inspection for detecting cracking. However, a crevice may be present near the weld and visual inspection may not be adequate for detecting cracks initiated in the crevice region. Provide justification for why augmented inspection technique, that includes ultrasonic testing (UT) or other demonstrated acceptable inspection method for the welded access hole cover (see NUREG 1801, Item IV.B1.1.4), is not required. Otherwise, provide augmented inspection as specified in NUREG 1801, Item IV.B1.1.4.

Response:

The augmented inspection technique discussed in NUREG 1801, Item IV.B1.1.4 is applicable to welded access hole covers. Dresden Unit 2 and Quad Cities Units 1 and 2 have replaced the welded access hole covers with mechanical covers. Therefore, the augmented inspections are not required on these units. The Dresden Unit 3 welded access hole covers are inspected visually and augmented by ultrasonic examination consistent with the requirements of GE SIL 462 "Shroud Access Cover Cracking and Radial Cracking" Revision 1, as specified in NUREG 1801, Item IV.B1.1.4. This inspection is specified in LRA Table 3.1-1, Ref. No.3.1.1.18.

RAI 3.1-9

(a) In Section 3.1.1.1.5 of the LRA, the applicant states that thermal stratification, thermal cycling and thermal stripping, thermal transients, and flow accelerated corrosion are potential aging mechanisms for small-bore piping. The LRA also states that a review of the Dresden and Quad Cities Risk Informed Inservice Inspection (RI-ISI) Evaluations on degradation mechanism assessment demonstrated that only Dresden had a high failure potential on a small bore pipe due to thermal fatigue. The inspection will consist of an ultrasonic exam on one of the two-inch drain lines off the Dresden main steam header. These lines are Class 1 and within the scope of License Renewal. The staff has the following comments:

- c Identify all Class 1, small bore piping in all Units (Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2). Include the pipe sizes, material and type of weld (i.e., butt or socket). If there are no UT-inspectable full penetration butt welds within scope, then socket welds that are replaced due to modifications should be destructively tested to confirm the effectiveness of the existing AMPs. This is consistent with NUREG-1801, Section XI.M32, which allows a plant-specific destructive examination of replaced piping in lieu of NDE that permits inspection of the inside surfaces of the piping.

- c As currently written, 10 CFR Part 54 does not allow the staff to accept the elimination of SSCs from aging management based on risk-informed arguments. Therefore, RI-ISI evaluations can be used to select susceptible SSCs locations, but can not eliminate SSCs from being inspected for a one-time inspection program. A sampling of butt welds from each unit should be developed, that is consistent with the ASME Code, and is sufficient to confirm the effectiveness of existing AMPs and/or to confirm that there is no need to manage aging-related degradation for the period of extended operation. Inspecting one weld, in one unit is not a sufficient sample size. Provide a sampling plan with a suitable sample size and an explanation of the selection process. This plan should also include a discussion regarding expansion of the inspection sample size and locations for follow up of unacceptable inspection findings as required by NUREG-1801, Section XI.M32. This plan is to be reviewed by the staff on a plant-specific basis, as required by NUREG-1801, Section XI.M32.

- c Section 3.1.1.1.5 of the LRA does not specify an inspection program for stress corrosion cracking (SCC) as an aging mechanism in small bore piping. What programs will be used to manage SCC in small bore piping?

(b) The applicant stated that, for this AMP, the one-time inspection program for small-bore Class 1 piping less than 4 inches will consist of an ultrasonic exam on one of the two-inch drain lines off the Dresden main steam header. These lines were identified as part of a review of the Dresden and Quad Cities Risk Informed Inservice Inspection (RISI) degradation mechanism assessments on Class 1 piping. The aging mechanisms cited by the report for these lines are thermal stratification, cycling, and stripping (TASCS), thermal transients (TT), and flow accelerated corrosion. Nuclear industry service experience, documented in several industry and NRC reports, has shown that the majority of reported piping leaks occur in small bore piping less then 4-inch NPS. A significant number of these failures have been reported in reactor coolant system, main steam system, feedwater system, and auxiliary systems in BWR plants. Also, a large portion of the reported Class 1 small bore piping failures occurred in piping 1-inch NPS and less that were caused primarily by mechanical vibration, thermal fatigue/turbulent penetration, stress corrosion cracking, and erosion-corrosion aging mechanisms. Since Class 1 small bore piping 1-inch NPS and less are exempt from NDE examinations in ASME Section XI, these lines will typically receive only periodic VT-2 visual examination. In addition, many RI-ISI evaluations do not include Class 1 piping 1-inch NPS and less in their evaluation scope and specific degradation mechanism assessments are not performed for these lines.

Therefore, it is not clear that the applicant's proposed one-time inspection program for small-bore piping will be representative of all Class 1 piping 1-inch NPS and less with full penetration butt welds (socket welds are excluded).

The applicant is requested to clarify whether the Dresden and Quad Cities Risk Informed Inservice Inspection (RI-ISI) degradation mechanism assessments included Class 1 piping 1-inch NPS and less with full penetration butt welds. Also describe how the proposed one-time inspection program will confirm that the aging mechanisms associated with the Class 1 small-bore piping 1-inch NPS and less with full penetration butt welds at Dresden and Quad Cites are either not occurring and/or there is no need to manage age-related degradation for the period of extended operation.

<u>RAI 3.1-10</u>

In LRA Section 3.1.1.1.6, the applicant states that the reactor vessel flange leak detection line at Quad Cities is a Class 2 stainless steel component, and is susceptible to cracking due to stress corrosion cracking and intergranular stress corrosion cracking. Quad Cities ISI Program, Relief Request PR-02 (relief granted per SER dated 9/15/95), provides for an alternate inspection of the reactor vessel flange leak detection line. This alternate examination utilizes a VT-2 visual examination once each inspection period on the line during vessel flood-up during a refueling outage. This alternate examination is not acceptable for license renewal since cracking can not be detected in the vessel flange leak detection line before its intended function is compromised. However, performance of VT-2 examination every refueling outage would be acceptable. A commitment to performing this alternate examination every refueling outage for license renewal is needed.

Response:

This RAI has been replaced by RAI 3.1-25 per email from Tae Kim, USNRC, dated September 8, 2003.

<u>RAI 3.1-11</u>

LRA sections 3.1.1.1.2 and 3.1.1.1.7 states that the heat exchanger test and inspection activities described in LRA Appendix B.2.6 will augment the ASME Section XI ISI program described in LRA Appendix B.1.1. LRA Sections 3.1.1.1.2 and 3.1.1.1.7 does not identify any augmented inspection to detect, loss of material, and crack initiation and growth in isolation condenser tubesheet, channel head, and shell as recommended by Items C1.4-a, and C1.4-b Chapter IV.C1 of NUREG-1801 respectively. LRA Appendix B.1.1 requires VT-2 examinations of the reactor coolant pressure boundary during system pressure testing. This is not adequate for detecting crack initiation and growth in the isolation condenser components before their intended function (pressure boundary) is compromised. Identify the augmented inspection program for detecting loss of material, and crack initiation and growth in the Dresden isolation condenser tubesheet, channel head, and shell as recommended by Items C1.4-a and C1.4-b, Chapter IV.C1 of NUREG-1801.

Response:

Aging management program B.2.6, Heat Exchanger Test and Inspection Program, manages loss of material and crack initiation and growth in the Dresden isolation condensers. This program, as it applies to the isolation condensers, which include the tubing, tube sheets, channel heads and shells, consists of performing eddy current testing of the tubes as well as temperature and radiation monitoring of the shell-side (cooling) water. These activities are consistent with the augmented activities recommended by NUREG-1801, items IV.C.1.4-a and b.

RAI 3.1-12

The applicant identifies loss of material due to general, pitting, and crevice corrosion as an applicable aging effect for reactor vessel, stainless steel valves, and carbon steel piping, fittings, and valves exposed to wet gas. Provide a description of the wet gas environment and evaluate its impact to cause general, pitting and crevice corrosion for reactor vessel, stainless steel valves, and carbon steel piping, fittings and valves.

Response:

The License Renewal Application identifies loss of material due to pitting and crevice corrosion as an applicable aging effect for stainless steel valves (LRA Table 3.1-2, Aging Management Reference 3.1.2.51), loss of material due to general, pitting and crevice corrosion for carbon steel piping and fittings (LRA Table 3.1-2, Aging Management Reference 3.1.2.51) and carbon steel valves exposed to wet gas (LRA Table 3.1-2, Aging Management Reference 3.1.2.48). The application does not identify the reactor vessel as being exposed to wet gas.

The wet gas environment is an air environment that contains moisture. There are three aging mechanisms associated with the loss of material caused by the wet gas environment: general corrosion, pitting corrosion, and crevice corrosion.

The following information is provided by EPRI 1003056 Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3.

- General corrosion is the result of a chemical or electrochemical reaction involving carbon steel in the presence of oxygen and moisture.

- Pitting corrosion is a form of localized attack. Pitting corrosion can occur at a gas-to-liquid interface, which is commonly called water-line attack. Both carbon steel and stainless steel are susceptible to pitting corrosion under a wet gas environment.

- Crevice corrosion occurs when a crevice exists in carbon or stainless steel that allows a corrosive environment to develop within the crevice. Oxygen and moisture in the present of contaminants can provide an environment severe enough to propagate crevice corrosion when subject to cyclic wet/dry conditions.

Material at the gas-to-fluid interface is susceptible to crevice corrosion due to the possible wetting/drying cycling.

RAI 3.1-13

The applicant does not identify loss of preload as an aging effect for the closure bolting in the reactor vessel system, recirculation pumps, reactor recirculation valves, reactor vessel head vent valves, and the reactor coolant pressure boundary portion of all other systems. In LRA Appendix B.1.12, the applicant states that loss of preload in a mechanical joint is a design driven process and, therefore, it is not an aging effect. Loss of preload, however, may take place during operation when closure bolting is subject to stress relaxation cyclic loads and differential thermal expansion. NUREG-1801, Chapter XI.M18, Bolting Integrity, requires this program to include periodic inspection of closure bolting for indication of loss of preload. Discuss why periodic inspection of those closure bolting for indication of loss of preload due to the aforementioned mechanisms is not required. If periodic inspection is required, reference the appropriate AMP and include the appropriate inspection in the AMP.

Response:

Upon further evaluation, Exelon will manage the loss of preload for closure bolting in the reactor vessel system, recirculation pumps, reactor recirculation valves, reactor vessel head vent valves, and the reactor pressure boundary portion of all other systems. Aging management program, B.1.12, Bolting Integrity, will be enhanced to include periodic inspections of the closure bolting in accordance with the ASME Code Section XI requirements. Closure bolting will be periodically inspected for signs of leakage. The enhanced Bolting Integrity aging management program will be comprised of periodic In-Service Inspection (ISI) and piping and component Preventive Maintenance inspections. These activities will detect early leakage and material degradation of closure bolting (that may be caused by loss of material or cracking) prior to loss of system or component intended functions. Periodic In-Service Inspection of closure bolting was accepted by the NRC staff as an acceptable aging management program for loss of pre-load for the components discussed above in NUREG-1769, Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, Section 3.1.3.2.1.

RAI 3.1-14

The applicant credits LRA Appendix B.1.1, ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD, for managing loss of fracture toughness due to thermal aging embrittlement in reactor recirculation system valve bodies and pump casings made of cast austenitic stainless steel, and reactor water cleanup valve bodies made of cast austenitic stainless steel. The inservice inspection program includes visual inspection for detecting cracks in the CASS valve bodies and pump casings.

(a) Explain how the proposed visual inspection technique is qualified for detecting IGSCC cracks in the CASS pump casings. Has Code Case N-481 been used to supplement the ISI requirements of ASME Code Section XI for these pump casings? While implementing this code case, was an flaw evaluation performed for this aging effect? If not, evaluate this as a TLAA in accordance with 10 CFR Part 54.3.

(b) Since ASME Section XI, Subsection IWB, provides little guidance as to how flaws detected in CASS components(valve bodies and pump casings) should be evaluated to determine acceptability for continued service, will NUREG-1801, XIM.12 acceptance criteria be met.

Response:

(a) NUREG-1801 was relied on as an approved topical report in the preparation of the LRA. As such, the recommendations from NUREG-1801, Lines IV.C1.2-c, and IV.C1.3-b were considered. These NUREG-1801 lines state, "For pump casings (and valve bodies), screening for susceptibility to thermal aging is not required. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS valve bodies." Therefore, no additional inspections are required. Code Case N-481 "Alternative Examination Requirements for Cast Austenitic Pump Casings" does not supplement the Dresden or Quad Cities ISI requirements.

(b) None of the NUREG-1801, Chapter IV line items recommend the implementation of NUREG-1801, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)", for the reactor recirculation system valve bodies and pump casings made of cast austenitic stainless steel or the reactor water cleanup valve bodies made of cast austenitic stainless steel. As stated in (a) above, the requirements of ASME Section XI are sufficient for managing the effects of loss of fracture toughness.

RAI 3.1-15

NUREG-1801 requires inspection and water chemistry as AMP's for stainless steel and CASS components (NUREG-1801, item IV.C1.1-f). The applicant credits only LRA Appendix B.1.2, "Water Chemistry," for managing cracking in these components (see Reference Nos. 3.1.2.26, 3.1.2.29, 3.1.2.40, 3.1.2.49, and 3.1.2.52, 3.1.2.25 and 3.1.2.53 in LRA Table 3.1-2). Compare the environments for components equivalent to NUREG-1801, items IV.C1.1-f to those identified in Reference Nos. 3.1.2.26, 3.1.2.29, 3.1.2.40, and 3.1.2.49, 3.1.2.52 (Recirculating System); 3.1.2.13, 3.1.2.23, 3.1.2.24, 3.1.2.38, 3.1.2.49 and 3.1.2.52(Nuclear Boiler Instrumentation system)in LRA Table 3.1-2. Provide basis for concluding Reference Nos. 3.1.2.26, 3.1.2.29, 3.1.2.40, 3.1.2.49, 3.1.2.52 (Recirculating System) and 3.1.2.13, 3.1.2.23, 3.1.2.24, 3.1.2.38, 3.1.2.49, 3.1.2.52 (Nuclear Boiler Instrumentation system) in LRA Table 3.1-2 does not require inspection.

Response:

Exelon has reviewed LRA, Table 3.1-2, Aging Management References 3.1.2.26, 3.1.2.29, 3.1.2.40, 3.1.2.49, 3.1.2.52 (Recirculating System) and 3.1.2.13, 3.1.2.23, 3.1.2.24, 3.1.2.38, 3.1.2.49, 3.1.2.52 (Nuclear Boiler Instrumentation system). The following technical explanation is provided for why these components do not require one-time inspections.

The piping components represented by these references are small-bore (2" and under) socket welded components, downstream of the excess flow check valves and located outside primary containment. The normal operating temperature is less than 140°F, the minimum temperature needed to initiate IGSCC. Therefore, the Aging Management References should have reported an environment of Reactor Coolant Water (< 140°F); Aging Effect/Mechanism as "None" and the Aging Management Program as "None."

Note that this environment was not part of the original License Renewal Application, nor was it contained in the response provided to RAI 3.0-1, submitted to the NRC on 6/11/03. This piping was originally considered to be in the 288°C (550°F) reactor water coolant environment, similar to the piping to which it is attached. However, the actual normal operating environment is <140°F.

Additionally, LRA Table 2.3.1-5, "Components Requiring Aging Management Review – Reactor Recirculation System," Component Group – NSR Vents and Drains, Piping and Valves (attached support), should not have indicated Dresden only.

RAI 3.1-16

The applicant identifies cumulative fatigue damage as an applicable aging effect only for the reactor head vent system valves but not for piping and fittings. Explain why cumulative fatigue damage is not an applicable aging effect for the reactor head vent system piping and fittings.

Response:

Cumulative fatigue damage is an applicable aging effect for the reactor head vent piping and fittings. LRA Table 2.3.1-6 should have included Aging Management Reference 3.1.1.1 for Component Group, "Piping and Fittings" as shown below.

Component	Component Intended Function	Aging Management Ref
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.2.4

RAI 3.1-17

According to Aging Management Review Aid for the reactor vessel head vent system (Table 2.3.1-6), the applicant identifies crack initiation and growth due to SCC and IGSCC as an applicable aging effect for the reactor head vent system austenitic stainless steel valve bodies exposed to reactor coolant water at 288 °C. The applicant, however, does not identify cracking as an applicable aging effect for the reactor head vent system CASS valve bodies exposed to reactor coolant water. The CASS valve bodies are susceptible to cracking due to IGSCC if its ferrite content is less than 7.5 vol.% and carbon content greater than 0.035 wt% and if they are exposed to BWR reactor coolant water at 288 °C. The applicant needs to explain why cracking is not an applicable aging effect for CASS valve bodies in reactor vessel head vent system. If cracking is an aging effect, then are there appropriate AMP's consistent with NUREG-1801,Section IV, Item C.1.1-f?

Response:

Dresden and Quad Cities do not have cast austenitic stainless steel (CASS) valves installed in the reactor vessel head vent system.

RAI 3.1-18

In LRA Section 3.1.1.2.2, the applicant states that the carbon steel components in the reactor vessel head vent system and the nuclear boiler instrumentation system are not susceptible to flow-accelerated corrosion because these components operate for less than 2% of the plant operating time or at flow rates less than 1.8 m/s (6 ft/s). The applicant references EPRI reports NSAC-202L-R2 and TR-114882 as the bases for these criteria. Chapter XI.M17, "Flow-Accelerated Corrosion," of NUREG-1801 only relies on EPRI report NSAC-202L-R2 for an effective FAC program. Does EPRI report NSAC-202L-R2 state that carbon steel components are not susceptible to FAC, and do not require aging management when these components are operated at flow rates less than 1.8m/s(6ft/s)? If not, then specify the applicable aging management program as required by NUREG-1801.

Response:

Exelon has reevaluated the use of EPRI TR-114882 and NSAC-202L-R2 and has decided not to take an exception to NUREG-1801 for aging management of the reactor vessel head vent system.

The reactor vessel head vent system will be included in the Dresden and Quad Cities flow accelerated corrosion program and LRA Section 3.1.1.2.2 should not have included the reactor vessel head vent system in the exception described in section 3.1.1.2.2 of the LRA.

The reactor vessel bottom head drain lines are included in the Dresden and Quad Cities flow accelerated corrosion program. Except for the reactor vessel bottom head drain lines, all components in the nuclear boiler instrumentation system are made of stainless steel, experience no flow, or operate less than 2% of plant operating time. Therefore, the reactor vessel bottom head drain lines are the only components in the nuclear boiler instrumentation system that are susceptible to flow accelerated corrosion.

LRA section 3.1.1.2.2 should have read as follows

> *Flow accelerated corrosion is an applicable aging mechanism for the main steam lines, feedwater lines, reactor vessel head vent lines and reactor vessel bottom head drain lines (evaluated with the nuclear boiler instrumentation system).*

However, carbon steel components in the core spray, shutdown cooling (Dresden only), HPCI, RCIC (Quad Cities only), nuclear boiler instrumentation (except for the reactor vessel bottom head drain lines) are not susceptible to flow accelerated corrosion and do not require aging management. This exception is based on the following:

1. EPRI NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program", allows an exclusion from flow-accelerated corrosion for systems with no flow or those that operate less than 2% of plant operating time.

2. NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal", states that erosion/corrosion in HPCI and RCIC turbine steam supply piping is non-significant due to the low flow range.

3. Dresden and Quad Cities operate these systems less than 2% of plant operating time. Additionally, plant experience has not revealed flow-accelerated corrosion in these lines.

The aging management results for managing flow-accelerated corrosion of the carbon steel components in the Reactor Vessel Head Vent and the Nuclear Boiler Instrumentation System are provided in LRA Table 3.1-1, Aging Management Reference 3.1.1.11.

No revision is required to LRA Table 2.3.1-6 (Component Groups Requiring Aging Management Review – Reactor Vessel Head Vents). Component Group "Piping and Fittings" and component Group "Valves" currently cite Aging Management Reference 3.1.1.11.

LRA Table 2.3.1-7 (Component Groups Requiring Aging Management Review - Nuclear Boiler Instrumentation) should have removed "Quad Cities only" from the Piping and Fittings line. LRA Table 2.3.1-7 should have read as follows.

Component	Component Intended Function	Aging Management Ref
Piping and Fittings	Pressure Boundary	3.1.1.11, 3.1.1.15, 3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.25, 3.1.2.26

<u>RAI 3.1-19</u>

According to Aging Management Review Aid for the reactor vessel head vent system (Table 2.3.1-6), the reactor head vent system includes CASS valve bodies exposed to 288 °C (550 °F) reactor coolant water. The applicant, however, does not identify the loss of fracture toughness due to thermal aging embrittlement as an applicable aging effect for these components. Explain why loss of fracture toughness is not considered for CASS valve bodies in the reactor vessel head vent system. If loss of fracture toughness is identified as an applicable aging effect, then provide a program for managing that effect.

Response:

Dresden and Quad Cities do not have cast austenitic stainless steel (CASS) valves installed in the reactor vessel head vent system.

The material for the line in Table 2.3.1-6 of the Aging Management Review Aid that shows valves with material of "Carbon Steel, Cast Austenitic Stainless Steel, Stainless Steel" should have read "Carbon Steel, Stainless Steel". Cast Austenitic Stainless should have been removed from the list of materials.

RAI 3.1-20

(a) The applicant does not identify the loss of fracture toughness due to thermal aging embrittlement as an applicable aging effect for CASS CRD valve bodies located around CRD housings in the nuclear boiler instrumentation system. Explain why loss of fracture toughness is not an applicable aging effect for these valve bodies. If loss of fracture toughness is identified as an applicable aging effect, then provide a program for managing that effect.

(b) The applicant identifies loss of fracture toughness due to thermal aging embrittlement as an applicable aging effect for CASS valve bodies in the reactor water cleanup system but not in the control rod drive hydraulic systems. Both of these systems are internally exposed to 288 □C (550 □F) reactor coolant water.

Explain why loss of fracture toughness is not an applicable aging effect for CASS valve bodies in the control rod drive hydraulic system.

Confirm whether there are any other reactor coolant pressure boundary components in the other systems that are made of CASS

If there are CASS PB components in the other systems, then submit AMR results for those components.

Response:

(a) NUREG 1801, Aging Management Program XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS), Scope of Program, states, "The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F)".

The valves associated with the nuclear boiler instrumentation system are located outside the drywell and are not insulated. The reactor coolant temperature through these valves is below 250°C (482°F). Additionally, the material for these valves is ASTM 182 not ASTM A351. The valves associated with the control rod drive hydraulic system are continuously supplied with cooling water < 38°C (100°F) from the cooling water header of the Control Rod Drive Hydraulic system. This maintains the control rod drives and all associated valve temperatures to less than 121°C (250°F).

For these reasons, fracture toughness is not an applicable aging effect.

(b) As discussed in response to question (a), valves in the CRD Hydraulic System are not exposed to temperatures of 250°C (482°F) or greater. Only the Reactor Water Cleanup System and the Reactor Recirculation System include stainless steel valves in this category. All other stainless steel valves with an operating temperature of less then 250°C (482°F) are assigned material types of stainless steel casting or stainless steel with only the aging effect of crack initiation and growth. The aging management results for CASS valves in the Reactor Water Cleanup System and the Reactor Recirculation System are provided in LRA Table 3.1-1, Aging Management References 3.1.1.9 and 3.1.1.15.

<u>RAI 3.1-21</u>

(a) In LRA Table 3.1-2 (Ref. No. 3.1.2.11, 3.1.2.26, 3.1.2.35, and 3.1.2.52), the applicant identifies crack initiation and growth due to SCC as an applicable aging effect for "stainless steel casting" valves, filters/strainers; and SS tanks, piping & fittings in the CRD hydraulic system exposed to oxygenated water up to 288 □C (550 □F). NUREG-1801, however, does not address aging management of these CRD components. Submit industry experience and plant-specific experience related to aging degradation of these CRD components. Based on experience, provide justification for not requiring inspection (Item 3.1.2.11 requires Water Chemistry).

(b) The applicant credits LRA Appendix B1.2, Water Chemistry Program only, for managing crack initiation and growth due to SCC for "stainless steel casting" valves, filters/strainers; and SS tanks and piping & fittings in the CRD hydraulic system exposed to oxygenated water up to 288 □C (550 □F). The staff notes that Appendix B1.2 is just a mitigative program and not a condition-monitoring program. Provide a program to verify the effectiveness of the water chemistry program.

Response:

(a) No incidents of crack initiation and growth due to stress corrosion cracking (SCC) in the control rod drive (CRD) system were identified at Dresden or Quad Cities.

A review of industry experience, including Dresden and Quad Cities, noted problems with degradation of the surface plating on CRD hydraulic accumulator interior surfaces resulting in some corrosion and pitting of the plated carbon steel. However, this degradation is not associated with stress corrosion cracking.

In addition, cracking was discovered in the CRD hydraulic control system return line near its connection to the reactor. The CRD return line to the reactor has been removed for both Dresden and Quad Cities, thereby eliminating this concern for SCC in the CRD system. Except for the return line to the reactor, the review of operating experience did not produce any indications of SCC in the CRD system. SSC has not occurred in the CRD system at Dresden or Quad Cities and the Dresden and Quad Cities CRD systems have been modified to remove the components where SSC has occurred at other BWRs. The Dresden and Quad Cities experience base, together with the CRD systems' modification, supports a conclusion that properly controlled water chemistry is adequate to eliminate the potential for SSC and inspection for occurrence of SSC in the CRD system is not required.

(b) SCC occurs through the combination of high stress (both applied and residual tensile stresses), a corrosive environment, and a susceptible material. Elimination or reduction in any of these three factors will decrease the likelihood of SCC occurring. The CRD System water is supplied by the Condensate Storage Tank (CST). The water in the CST is monitored and controlled to keep known detrimental contaminants below the system specific limits indicated in the EPRI water chemistry guidelines (TR-103515) to mitigate corrosion.

The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The NUREG 1801 identifies circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, control of CST chemistry in accordance with EPRI guidelines does not preclude loss of material of stainless steel at locations of stagnant flow conditions. Accordingly in those cases, verification of the effectiveness of the CST chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the period of extended operation. As discussed in NUREG 1801, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system. Aging Management Program B.1.23, "One-Time Inspection", requires an inspection of components exposed to CST water. An inspection is to be conducted of stainless steel CRD components exposed to CST water to verify the effectiveness of CST chemistry and confirm the absence of loss of material in stagnant flow areas as required by NUREG 1801. The references for these inspections are in LRA Table 3.1-2, Aging Management References 3.1.2.10, 3.1.2.25, 3.1.2.34 and 3.1.2.53. Water chemistry controls that are sufficiently effective to prevent loss of material at stagnant flow locations are also expected to be effective at preventing stress corrosion cracking.

<u>RAI 3.1-22</u>

The applicant identifies loss of material due to wear as an applicable aging effect for closure bolting in the SBLC system but not in the HPCI, core spray, RCIC, RHR, LPCI, RWCU, MS, and FW systems, and the isolation condenser externally exposed to air or nitrogen with metal temperature up to 288 □C (550 □F). Provide technical basis for not identifying loss of material due to wear as an applicable aging effect for the closure bolting in the reactor coolant pressure boundary portion of all the other systems except SBLC system.

Response:

The LRA does identify loss of material due to wear as an applicable aging effect for closure bolting in the HPCI, core spray, RCIC, RHR and LPCI systems and in the isolation condenser. Loss of material due to wear is identified for closure bolting in the following LRA Chapter 2.3 tables and is managed by the Bolting Integrity aging management program B.1.12.

Chapter 2 Table	System	Aging Management Ref
2.3.2-1	High Pressure Coolant Injection System	3.2.2.4
2.3.2-2	Core Spray System	3.2.2.4
2.3.2-3	Containment Isolation Components and Primary Containment Piping System	3.2.2.4
2.3.2-4	Reactor Core Isolation Cooling System	3.2.2.4
2.3.2-5	Isolation Condenser	3.2.2.4
2.3.2-6	Residual Heat Removal System	3.2.2.4
2.3.2-7	Low Pressure Coolant Injection System	3.2.2.4
2.3.2-8	Standby Liquid Control	3.1.2.2

Loss of material due to wear and crack initiation and growth due to cyclic loading should have been identified for closure bolting in systems RWCU, MS and FW and will be managed with the Bolting Integrity aging management program B.1.12.

The environment for closure bolting Aging Management References 3.1.2.1, 3.1.2.2, 3.2.2.1 and 3.2.2.4 should have read "Air with metal temperature up to 288°C (550°F)". This environment description is consistent with the environment used in NUREG-1801 Chapter IV Item C1.3-e and C1.3-f

LRA Tables for RWCU, MS and FW should have read as follows:

Table 2.3.3-4 Component Groups Requiring Aging Management Review – Reactor Water Cleanup System

Component Group	**Component Intended Function**	**Aging Management Ref**
Closure Bolting	Pressure Boundary	3.1.1.1, 3.1.1.12, 3.1.2.1, 3.3.1.22

Table 2.3.4-1 Component Groups Requiring Aging Management Review - Main Steam

Component Group	**Component Intended Function**	**Aging Management Ref**
Closure Bolting	Pressure Boundary	3.1.1.1, 3.1.1.12, 3.1.2.1, 3.4.1.6

Table 2.3.4-2 Component Groups Requiring Aging Management Review - Feedwater System

Component Group	**Component Intended Function**	**Aging Management Ref**
Closure Bolting	Pressure Boundary	3.1.1.1, 3.1.1.12, 3.1.2.1, 3.4.1.6

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.1	Closure Bolting	Low-Alloy Steel	Air with metal temperature up to 288°C (550°F)	Crack initiation and growth/ Cyclic loading	Bolting Integrity (B.1.12)	NUREG-1801 does not address Crack initiation and growth/ Cyclic loading for BWR closure bolting.
3.1.2.2	Closure Bolting	Low-Alloy Steel	Air with metal temperature up to 288°C (550°F)	Loss of material/ Wear	Bolting Integrity (B.1.12)	Consistent with Table 3.1-1 Aging Reference 3.1.1.12

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.1	Closure Bolting	Low-Alloy Steel	Air with metal temperature up to 288°C (550°F)	Crack initiation and growth/ Cyclic loading	Bolting Integrity (B.1.12)	NUREG-1801 does not address Crack initiation and growth/ Cyclic loading for BWR closure bolting.
3.2.2.4	Closure Bolting	Low-Alloy Steel	Air with metal temperature up to 288°C (550°F)	Loss of material/ Wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address Loss of material/ Wear for BWR closure bolting.

RAI 3.1-23

D/QCNPS has implemented extended power uprates to increase the power output of each of the four units. Such uprates are often accompanied by increases in main steam and feedwater flows in BWRs. Explain how the effects of extended power uprates are taken into account in identifying components susceptible to wall thinning due to flow-accelerated corrosion.

Response:

At pre-EPU conditions the steam flow and the feedwater flow at Dresden and Quad Cities were –

Parameter	Dresden	Quad Cities
Vessel Steam Flow (Mlbm/hr)	9.81	9.76
Feedwater Flow (Mlbm/hr)	9.78	9.73

At the licensed EPU conditions the steam flow and the feedwater flow at Dresden and Quad Cities are –

Parameter	Dresden	Quad Cities
Vessel Steam Flow (Mlbm/hr)	11.71	11.71

Parameter	Dresden	Quad Cities
Feedwater Flow (Mlbm/hr)	11.68	11.68

These values correspond to increases of approximately 20% in both steam flow and feedwater flow rates.

The increases in steam flow and feedwater have been considered and appropriately incorporated into the flow accelerated corrosion (FAC) programs at Dresden and at Quad Cities. The predictive analysis, CHECWORKS, has been updated to reflect uprate design conditions such as mass flow, temperature, and steam quality. Where appropriate, inspection intervals have been moved forward to address increased wear rates.

RAI 3.1-24

The applicant credits BWR stress corrosion cracking (LRA Appendix B.1.7) and water chemistry (LRA Appendix B.1.2) for managing crack initiation and growth due to SCC and IGSCC in SS components in the HPCI, core spray, RCIC, RHR, LPCI, SBLC, SDC, RWCU, MS, and FW systems and the isolation condenser. The applicant also states that the BWR stress corrosion cracking AMP is based on BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules.

(a) Describe plant-specific experience related to IGSCC cracking of the SS components in the HPCI, core spray, RCIC, RHR, LPCI, SBLC, SDC, RWCU, MS, and FW systems and the isolation condenser.

(b) Submit information on the mitigation actions taken at D/QCNPS with respect to selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stress and monitoring of water chemistry as specified by NUREG-1801, Chapter XI.M7.

(c) Confirm whether hydrogen water chemistry and noble metal chemical application (NMCA) are implemented at D/QCNPS. If so, explain how this implementation has affected monitoring of water chemistry parameters.

(d) Submit information on the inspection frequency (based on whether hydrogen water chemistry and/or noble metal chemical applicator are used) and the corresponding number of welds to be inspected following the BWRVIP-75 guidelines.

Response:

(a) Exelon has reviewed the Dresden and Quad Cities operating experience related to IGSCC of stainless steel components in the following station systems. Although not requested, the reactor recirculation system was included because of its extensive history.

- Main Steam (MS)
- Reactor Water Cleanup (RWCU)
- Feedwater (FW)
- Core Spray

- Control Rod Drive (CRD)
- Isolation Condenser (Dresden only)
- Low Pressure Coolant Injection (LPCI – Dresden only)
- High Pressure Coolant Injection (HPCI)
- Reactor Core Isolation Cooling (RCIC - Quad Cities only)
- Shutdown Cooling (SDC)
- Standby Liquid Control (SLBC)
- Residual Heat Removal (RHR)
- Reactor Recirculation

Reactor coolant pressure boundary piping was identified to have flaw indications such as indications on the reactor vessel safe ends and IGSCC on recirculation piping. However, there were no flaw indications (IGSCC) identified that affected the component intended function for any components in the above-mentioned systems. The following are representative examples of IGSCC operating experience related to reactor coolant pressure boundary piping. These examples demonstrate the effectiveness of the AMP.

- The IGSCC inspection of Quad Cities Unit 1 refueling outage Q1R15 in December 1998 identified some flaw indications on recirculation system piping that exceeded allowable values. Flaw evaluation and repair (weld overlay) were performed to justify continued plant operation. The recirculation piping is original piping and the associated IGSCC susceptible welds have received Induction Heat Stress Improvement (IHSI) treatment. The evaluation of the effectiveness of IHSI treatment for susceptible welds resulted in an adjustment of the inspection plan to change all Unit 1 28" IHSI treated Category C (non-resistant material, stress improvement after 2 years of unit operation) welds to Category D (non-resistant material, no stress improvement).

- IGSCC inspection during Dresden Unit 2 refueling outage D2R14 in June 1995 found circumferential crack indications in two recirculation pipe welds. Flaw evaluations were performed to support continued plant operation without repair. The inspection plan was adjusted to re-inspect these welds every refueling outage.

- During Dresden Unit 2 refueling outage D2R02 in February 1972, RPV safe ends were inspected by VT, PT and UT. Several indications were identified and ground out until they were acceptable.

- During Dresden Unit 2 refueling outage D2R05 in September 1977, 27 sensitized safe ends were inspected. On Nozzle N9 (CRD return nozzle), the indications were determined to be unacceptable and the safe end was replaced with 316L. This replacement was one of a number of safe end replacements made for Unit 2 (reference UFSAR Table 5.2-4) prior to the issuance of GL 88-01. The indication on Nozzle N2C (recirculation system inlet nozzle) was also unacceptable. It was ground out and a subsequent PT test was performed with acceptable results.

(b) The following mitigation actions are taken with respect to material selection, use of special process, and monitoring of water chemistry.

Dresden and Quad Cities have replaced the RWCU system piping with piping that is resistant to intergranular stress corrosion cracking (IGSCC).

- Dresden Unit 3 has replaced the reactor recirculation system piping with piping that is resistant (316 NG with maximum contents of 0.02 wt % carbon and 0.10 wt % nitrogen) to IGSCC.

- Replacement stainless steel components are provided in the solution annealed condition, with carbon less than 0.035 wt % and ferrite levels greater than 7.5 wt %.

- Existing stainless steel weldments are treated with IHSI to minimize tensile stresses and provide mitigation of IGSCC. Alloy 82 is used for nickel base alloy filler material.

- The Noble Metal Chemical (NMC) Injection system was installed to enhance the IGSCC mitigation. Decomposition of the compounds forms a thin layer (~1 $\mu g/cm^2$) of Pt and Rh, providing a catalytic surface on the reactor piping and internals. NMC injections have occurred at each site during outages (1999/2000 timeframe). No information is yet available on the effectiveness of the injections on IGSCC mitigation.

- The Hydrogen Water Chemistry (HWC) Injection system was installed to enhance the IGSCC mitigation by reducing the amount of oxidizing radiolysis products through the injection of hydrogen into feedwater while maintaining the concentration of reactor coolant ionic impurities. HWC injection is conducted continuously (as much as practicable) during normal unit operation. Data from Dresden Unit 2 has indicated that IGSCC in the reactor recirculation piping can be suppressed by HWC injection along with control of impurity concentrations in reactor water.

(c) Both Dresden and Quad Cities have implemented hydrogen water chemistry and noble metal chemical injection. As part of the implementation of HWC and NMC, monitoring of electrochemical corrosion potential (ECP) was added. This is a measure of the voltage that arises when a metal is in contact with a solution. ECP is measured by comparison to a standard reference electrode at the temperature of interest. ECP data and HWC index results are used to calculate crack growth rate factors of improvement. BWRVIP model for BWR crack growth indicates decreasing crack growth rate with decreasing ECP.

(d) At Quad Cities Station, Category C through E welds (Quad Cities currently has no Category B welds) are still being inspected to the same frequency as specified in BWRVIP-75, "BWR Vessel and Internal Project Technical Basis for Revisions of Generic Letter 88-01 Inspection Schedules," guidelines for normal water chemistry. Category A (resistant material) welds are fabricated with IGSCC resistant materials and are inspected per the RISI guideline. Quad Cities has implemented HWC and NMCA addition. However, it has not reduced the inspection frequencies as specified in BWRVIP-75 and NRC SER of EPRI Report TR-113932 ("BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)"), dated May 14, 2002. Based on the effectiveness of the HWC and NMCA, the inspection frequencies may be adjusted in the future, which will follow the requirements of BWRVIP-75.

At Dresden Station, Category C through E (cracked, overlay, or stress improved) welds (Dresden currently has no Category B welds) are being inspected to the frequency specified in BWRVIP-75 guidelines for normal water chemistry (NWC) or HWC/NMCA. Hydrogen water chemistry/noble metal chemical application inspection frequencies were reduced for Unit 2 and only applied to those weld locations where the improved water chemistry is effective. Unit 3 maintains the normal inspection frequencies. The hydrogen water chemistry system has been in use for Unit 2 since 1983. The system was not implemented for Unit 3 until 1996. Category A welds are fabricated with IGSCC resistant materials and are inspected per the RISI guidelines. There are no Category E welds on Dresden Unit-3.

The corresponding number of welds and frequency of inspection for Dresden Units 2 and 3 are provided in the following table.

Category	Total Population	Welds Inspected
Unit-2:		
C-HWC	28	3 (10% every 10 years)
C-HWC	66	17 (25% every 10 years)
D-HWC	41	41 (100% every 10 years, 50% in first six years)
D-HWC	24	24 (10% every 6 years)
E-HWC	37	4 (10% every 10 years)
E-HWC	1	1 (25% every 10 years)
Unit-3:		
C-NWC	50	13 (25% every 10 years)
D-NWC	13	13 (100% every 6 years)

RAI 3.1-25

In LRA Section 3.1.1.1.6, the applicant states that the reactor vessel flange leak detection line at Quad Cities is a Class 2 stainless steel component, and is susceptible to cracking due to stress corrosion cracking and intergranular stress corrosion cracking. Quad Cities ISI Program, Relief Request PR-02 (relief granted per SER dated 9/15/95), provides for an alternate inspection of the reactor vessel flange leak detection line through the 3rd ISI interval. This alternate examination utilizes a VT-2 visual examination on the line during vessel flood-up during a refueling outage. Future relief requests may be submitted by the applicant in accordance with 10 CFR 50.55a. Otherwise, the applicant must comply with the appropriate requirements of ASME Section XI. Please confirm that the aforementioned aging effects for the reactor vessel flange leak detection line at Quad Cities will be monitored/managed in accordance with the requirements of ASME Section XI, Table IWC-2500-1 for license renewal.

Response:

Quad Cities submitted Relief Request I4R-05, Quad Cities ISI Program (Ref. Letter SVP 2003-008, dated 1/17/03), which provides for an alternate inspection of the reactor vessel flange leak detection line through the 4th ISI interval. This alternate examination utilizes a VT-2 visual examination on the line during vessel flood-up during one refueling outage each inspection period. Quad Cities will manage the aging effects for the reactor vessel flange leak detection lines in accordance with the requirements of ASME Section XI, Table IWC-2500-1 as amended by NRC approved relief request(s) in accordance with 10 CFR 50.55a.

RAI 4.2.1

(a) In LRA Section 4.2-1, the applicant states that it has performed one bounding 54-EFPY fluence calculation for Dresden and one for Quad Cities and then used that fluence for determining corresponding 54-EFPY 1/4T fluence. Therefore, it is expected that the applicant used the same 54-EFPY 1/4T fluence for limiting beltline plate and weld material at both Dresden units. However, the data presented in Tables 4.2.1-1 through 4.2.1-4 indicate that the applicant has used two different values for the limiting beltline materials for Dresden; a fluence value of 3.9×10^{17} n/cm^2 for limiting plate and weld at Unit 1 and for limiting plate at Unit 2, and a value of 2.9×10^{17} n/cm^2 for limiting weld at Unit 2. The similar apparent discrepancy is present in LRA Tables 4.2.1-5 through 4.2.1-8 for Quad Cities. There appears to be another discrepancy between the peak fluence data for Quad Cities in LRA Sections 4.2.1 and 4.2.2. LRA Tables 4.2.1-5 through 4.2.1-7 for Quad Cities list 2.9×10^{17} n/cm^2 as the 54-EFPY 1/4T fluence, whereas LRA Table 4.2.2-2, also for Quad Cities, lists 3.9×10^{17} n/cm^2 as the 54-EFPY 1/4T fluence. A similar discrepancy is present between LRA Section 4.2.1 and 4.2.2 for 1/4T fluence data for Dresden. Explain these apparent discrepancies and provide revised Tables, as appropriate.

(b) The data for Cu content in the limiting beltline plate and weld material presented in LRA Section 4.2.1 appear to be different from the one presented in Appendix F of Dresden UFSAR. For example, LRA table 4.2.1-2 lists 0.24% Cu for Dresden Unit 2 limiting beltline weld material, whereas UFSAR Table 22 in Appendix F lists maximum Cu content of 0.21% for Dresden Unit 2. Resolve this apparent discrepancy.

(c) Provide all fluence data for all welds and plates in the beltline and specify which one is bounding in determining the USE.

(d) The applicant states that the 54-EFPY USE values reported in LRA Tables 4.2.1-1 through 4.2.1-8 will be managed in conjunction with the surveillance capsule results from the BWRVIP integrated surveillance program. The applicant also needs to include this commitment in the UFSAR supplements for Dresden and Quad Cities, LRA Appendix A.3.1.1, "Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement."

Response:

(a) The statement in the submittal means that one neutron transport (flux) calculation was prepared that bounds both Dresden and Quad Cities. However, based upon the different operating bases for the four units with regard to the time period of operation at different power levels, a unit-specific fluence was calculated for each of the four units. Calculations performed for each plant are provided below; the first term of each equation represents pre-EPU, followed by EPU. From these calculations it can be seen that, using the bounding flux with the plant-specific pre- and post-EPU periods of operation, the fluence at 54 EFPY is the same for all four units, when rounding is applied. The peak fluence on the vessel is located at approximately 82 inches above the bottom of active fuel. Additionally, axial flux distribution factors are applied to different elevations (by shell) in the beltline region. The peak fluence shown below is applied to the lower-intermediate shell and axial welds. For the lower shell, the peak fluence is adjusted by the axial flux distribution factor based on an elevation approximately 42 inches above the bottom of active fuel, which represents the lower to lower-intermediate girth weld. The axial factor of 0.71, at this location, is applied for pre-EPU; the lower to lower-intermediate girth weld, and all lower shell materials.

Dresden 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (19.4/54) + 3.46e8 n/cm2-s * 1.7e9 s * (34.6/54) = 5.67e17 n/cm2

Dresden 3 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (19.8/54) + 3.46e8 n/cm2-s * 1.7e9 s * (34.2/54) = 5.67e17 n/cm2

Quad Cities 1 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (21.1/54) + 3.46e8 n/cm2-s * 1.7e9 s * (32.9/54) = 5.66e17 n/cm2

Quad Cities 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (21/54) + 3.46e8 n/cm2-s * 1.7e9 s * (33/54) = 5.66e17 n/cm2

<u>Tables 4.2.1-1 through 4.2.1-8</u>: In calculating the USE % decrease for the limiting beltline material (plate or weld) of each unit, a combination of the applied fluence and the %Cu of each material is considered. In the case of Dresden 2, both the plate and weld limiting materials occur in the lower-intermediate shell, thereby using the same fluence. In the case of Dresden 3, the plate material also occurs in the lower-intermediate shell, thereby using the same fluence as that used for the Dresden 2 materials. However, the Dresden 3 limiting weld material with respect to the limiting USE % decrease occurs in the lower to lower-intermediate girth weld due to the higher copper content, which offsets the higher fluence and lower copper content of the materials in the other shells. This material sees a different (and lower) fluence than the lower-intermediate shell materials. A similar situation exists for Quad Cities 1. The plate and weld limiting materials with respect to the USE % decrease occur in the lower shell, where the fluence is lower. Similarly in Quad Cities 2, this situation exists for the plate material, while the limiting USE % decrease for the weld material occurs in the lower-intermediate shell where the fluence is higher.

<u>Sections 4.2.1 and 4.2.2 Fluence</u>: The values presented in LRA Tables 4.2.2-1 and 4.2.2-2 represent the PEAK fluence, both at the surface and at the 1/4T

locations. As noted above, an axial flux distribution factor is applied to the lower shell, thereby reducing the fluence (both peak and 1/4T) for the associated materials. Further, the values for ΔRT_{NDT} and ART provided in these tables represent the limiting materials that are based upon the fluence values presented.

(b) For the beltline region, Table 21 (Shell Course 57 – Lower Shell) and Table 22 (Shell Course 58 – Lower-Intermediate Shell) of the FSAR define actual chemical analysis for these materials. Tables 21 and 22 contain the chemical analysis for electroslag welds contained in the original FSAR. Since the original publication of the FSAR, the accepted best estimate chemistry for Electroslag Weld materials used in B&W vessels accepted by the NRC staff is 0.24% Cu and 0.37% Ni. "Evaluation of RT_{NDT}, USE and Chemical Composition of Core Region Electroslag Welds for Dresden Units 2 and 3", BAW-2258, Framatome Technologies, January 1996, provides a best estimate chemistry and initial RT_{NDT} for ESW materials, previously accepted by the NRC in P-T Curve report GE-NE-B13-02057-04R1a. Exelon submitted reactor vessel chemistry to the NRC in July 1998 in response to Generic Letter 92-01, Supplement 1. The information provided in that response is included in NRC database RVID2.

(c) The following tables list each beltline weld or plate for each unit, including the applied fluence at peak and 1/4T as well as the resulting 54 EFPY ART. The limiting material for each unit is highlighted in bold text.

Dresden and Quad Cities Weld Materials

Unit	Weld Heat or ID	Shell	54 EFPY Surface Fluence ($x10^{17}$ n/cm^2)	54 EFPY 1/4T Fluence ($x10^{17}$ n/cm^2)	54 EFPY ART (°F)	Bounding Material
Dresden 2	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	1P0661/8304	Low-Int	5.7	3.9	92	
	1P0815/8350	Low-Int	5.7	3.9	84	
	ESW	Lower	4.2	2.9	93	
	1P0815/8304	Lower	4.2	2.9	74	
	71249/8504	Lower to Low-Int (Girth)	4.2	2.9	82	
Dresden 3	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding***
	ESW	Lower	4.1	2.9	93	
	299L44/8650	Lower to Low-Int (Girth)	4.1	2.9	104	
Quad Cities 1	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	ESW	Lower	4.1	2.9	93	
	72445/8688	Lower to Low-Int (Girth)	4.1	2.9	81	
	406L44/8688	Lower to Low-Int	4.1	2.9	90	

		(Girth)				
Quad Cities 2	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	ESW	Lower	4.1	2.9	93	
	S3986/3870 Linde 124	Lower to Low-Int (Girth)	4.1	2.9	-3	

* The ART for the Dresden 3 ESW material is 0.4°F less than that for the girth weld material. However, using Code Case N-588, which allows a different application of K_I for girth weld materials, the limiting material for Dresden 3 is the ESW material(this is explained in more detail in the response to in RAI 4.2.2 (c)).

Dresden and Quad Cities Plate Materials

Unit	Plate Heat	Shell	54 EFPY Surface Fluence ($x10^{17}$ n/cm^2)	54 EFPY 1/4T Fluence ($x10^{17}$ n/cm^2)	54 EFPY ART (°F)	Bounding Material
Dresden 2	A9128-2	Lower	4.2	2.9	71	
	B3990-2	Lower	4.2	2.9	66	
	A9128-1	Lower	4.2	2.9	91	
	B4065-1	**Low-Int**	**5.7**	**3.9**	**95**	**Bounding**
	B5764-1	Low-Int	5.7	3.9	43	
	B4030-1	Low-Int	5.7	3.9	78	
	B4030-2	Low-Int	5.7	3.9	70	
Dresden 3	C1256-2	Lower	4.1	2.9	21	
	B5159-2	Lower	4.1	2.9	65	
	C1182-2	Lower	4.1	2.9	73	
	A0237-1	**Low-Int**	**5.7**	**3.9**	**83**	**Bounding**
	B5118-1	Low-Int	5.7	3.9	81	
	C1290-2	Low-Int	5.7	3.9	63	
Quad Cities 1	**B5524-1**	**Lower**	**4.1**	**2.9**	**72**	**Bounding**
	A0610-1	Lower	4.1	2.9	41	
	C1485-2	Lower	4.1	2.9	55	
	C1505-2	Low-Int	5.7	3.9	59	
	C1498-2	Low-Int	5.7	3.9	41	
	A0931-1	Low-Int	5.7	3.9	29	
Quad Cities 2	**C1516-2**	**Lower**	**4.1**	**2.9**	**52.1**	**Bounding**
	C1501-2	Lower	4.1	2.9	43	
	C1722-2	Lower	4.1	2.9	51	
	C2753-2	Low-Int	5.7	3.9	36	
	C2868-1	Low-Int	5.7	3.9	36	
	C3307-2	Low-Int	5.7	3.9	52.0	

(d) Section 4.2.2.2 of NUREG 1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, states that "The description (FSAR Supplement) should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1)." The UFSAR supplements for Dresden and Quad Cities LRA Appendix A.3.1.1, "Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement," contain sufficient information that satisfies section 4.2.2.2 of NUREG 1800. As such, no changes will be made. However, the changes requested by the NRC staff are already contained in section A.1.22, "Reactor Vessel Surveillance" of the UFSAR supplements contained in the LRA for both sites.

RAI 4.2.2

(a) In LRA Section 4.2.2, the applicant provides the results of one bounding calculation: the 54-EFPY peak surface fluence of 5.7×10^{17} n/cm^2 and peak 1/4T fluence of 3.9×10^{17} n/cm^2 for all four D/QCNPS vessels. Explain how you determined that the weld in which you calculated the neutron fluence bounds all the other welds in Dresden/Quad Cities.

(b) Using the calculated 54-EFPY peak 1/4T fluence, the applicant determines the 54-EFPY RT_{NDT} and ART values for all the beltline materials according to RG 1.99, Rev. 2. Out of all the 54-EFPY ART values, the applicant identifies the limiting ART value and lists it in LRA Tables 4.2.2-1 and 4.2.2-2 as the 54-EFPY ART for both Dresden and Quad Cities. Provide the 54-EFPY RT_{NDT} and ART values along with initial RT_{NDT} values for all the beltline materials for the four D/QCNPS reactor vessels.

(c) In LRA Section 4.2.2, the applicant states that due to the refinement in the approved methodology used to calculate the 54-EFPY fluence, the material with the limiting ART is the axial weld; with the exception of Dresden Unit 3 where the axial weld and girth weld ART values are identical. Identify the refinement mentioned here and explain how does it make the axial weld as a material having the limiting ART. The applicant invokes Code Case N-588 for Dresden Unit 3 that the causes axial weld to become the limiting material. Explain how the use of Code Case N-588 makes the axial weld the limiting material for Dresden Unit 3.

Response:

(a) One neutron transport (flux) calculation was prepared that bounds both Dresden and Quad Cities. However, based upon the different operating bases for the four units with regard to the time period of operation at different power levels, a unit-specific fluence was calculated for each of the four units. Calculations performed for each plant are provided below; the first term of each equation represents pre-EPU, followed by EPU. From these calculations it can be seen that, using the bounding flux with the plant-specific pre- and post-EPU periods of operation, the fluence at 54 EFPY is the same for all four units, when rounding is applied. The peak fluence on the vessel is located at approximately 82 inches above the bottom of active fuel. Additionally, axial flux distribution factors are applied to different elevations (by shell) in the beltline region. The peak fluence shown below is applied to the lower-intermediate shell and axial welds. For the lower shell, the peak fluence is adjusted by the axial flux distribution factor based on an elevation approximately 42 inches above the bottom of active fuel, which represents the lower to lower-intermediate girth weld. The axial factor of 0.71, at this location, is applied for pre-EPU; the lower to lower-intermediate girth weld, and all lower shell materials.

Dresden 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (19.4/54) + 3.46e8 n/cm2-s * 1.7e9 s * (34.6/54) = 5.67e17 n/cm2

Dresden 3 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (19.8/54) + 3.46e8 n/cm2-s * 1.7e9 s * (34.2/54) = 5.67e17 n/cm2

Quad Cities 1 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (21.1/54) + 3.46e8 n/cm2-s * 1.7e9 s * (32.9/54) = 5.66e17 n/cm2

Quad Cities 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm2-s * 1.7e9 s * (21/54) + 3.46e8 n/cm2-s * 1.7e9 s * (33/54) = 5.66e17 n/cm2

Sections 4.2.1 and 4.2.2 Fluence: The values presented in LRA Tables 4.2.2-1 and 4.2.2-2 represent the PEAK fluence, both at the surface and at the 1/4T locations. As noted above, an axial flux distribution factor is applied to the fluence at the lower shell, thereby reducing the fluence (both peak and 1/4T) for the associated materials. Further, the values provided in these tables for ΔRT_{NDT} and ART represent the limiting materials that are based upon the fluence values presented.

The following table lists each beltline weld for each unit, including the applied fluence at peak and 1/4T as well as the resulting 54 EFPY ART. The limiting material for each unit is highlighted in bold text.

Dresden and Quad Cities Weld Materials

Unit	Weld Heat or ID	Shell	54 EFPY Surface Fluence ($x10^{17}$ n/cm^2)	54 EFPY 1/4T Fluence ($x10^{17}$ n/cm^2)	54 EFPY ART (°F)	Bounding Material
Dresden 2	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	1P0661/8304	Low-Int	5.7	3.9	92	
	1P0815/8350	Low-Int	5.7	3.9	84	
	ESW	Lower	4.2	2.9	93	
	1P0815/8304	Lower	4.2	2.9	74	
	71249/8504	Lower to Low-Int (Girth)	4.2	2.9	82	
Dresden 3	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding***
	ESW	Lower	4.1	2.9	93	
	299L44/8650	Lower to Low-Int	4.1	2.9	104	

		(Girth)				
Quad Cities 1	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	ESW	Lower	4.1	2.9	93	
	72445/8688	Lower to Low-Int (Girth)	4.1	2.9	81	
	406L44/8688	Lower to Low-Int (Girth)	4.1	2.9	90	
Quad Cities 2	**ESW**	**Low-Int**	**5.7**	**3.9**	**104**	**Bounding**
	ESW	Lower	4.1	2.9	93	
	S3986/3870 Linde 124	Lower to Low-Int (Girth)	4.1	2.9	-3	

* The ART for the Dresden 3 ESW material is 0.4°F less than that for the girth weld material. However, using Code Case N-588, which allows a different application of K_I for girth weld materials, the limiting material for Dresden 3 is the ESW material (this is explained in more detail in RAI 4.2.2 (c)).

(b) Please see attached 54 EFPY ART tables at the end of this response for all four units.

(c) As can be seen in the attached ART tables, the ART for the Dresden 3 girth weld material is 104.16°F and the ART for the axial ESW material is 103.8°F. The detailed explanation and calculated basis for the use of the ESW material as the limiting material is provided in GE-NE-0000-0002-9600-01a, Revision 0 and is explained herein. Because the calculated value of K_{Im} is reduced for a girth weld due to the implementation of Code Case N-588 (circumferentially oriented defect for a circumferential weld), the axial weld bounds the P-T curve beltline region requirements. To demonstrate that by using Code Case N-588 the axial weld has the most limiting temperature for the P-T curves in the beltline region, the stress intensity calculations for both axial and girth welds at 54 EFPY are presented. It can be seen in the results of the axial and girth weld calculations that the "T" value for the axial weld (146.5°F) bounds the value for the girth weld (51°F). The following is an excerpt from the noted report.

Axial Weld Calculation:

The value of M_m for an inside axial postulated surface flaw from Paragraph G-2214.1 [6] was based on a thickness of 6.125 inches (the minimum thickness without cladding); hence, $t^{1/2} = 2.47$. The resulting value obtained was:

$$M_m = 1.85 \text{ for } \sqrt{t} \leq 2$$
$$M_m = 0.926 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464 = \mathbf{2.29}$$
$$M_m = 3.21 \text{ for } \sqrt{t} > 3.464$$

The stress intensity factor for the pressure stress is $K_{Im} = M_m \cdot \sigma$. The stress intensity factor for the thermal stress, K_{It}, is calculated as described in Section 4.3.2.2.4 except that the value of "G" is 20°F/hr instead of 100°F/hr.

Equation 4-9 can be rearranged, and 1.5 K_{Im} substituted for K_{IC}, to solve for (T - RT_{NDT}). Using the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [17], K_{Im} = 52.96, and K_{It} = 2.29 for a 20°F/hr coolant heatup/cooldown rate with a vessel thickness, t, that includes cladding:

$$\begin{aligned}(T - RT_{NDT}) &= \ln[(1.5 \cdot K_{Im} + K_{It} - 33.2) / 20.734] / 0.02 \\ &= \ln[(1.5 \cdot 52.96 + 2.29 - 33.2) / 20.734] / 0.02 \\ &= 42.5°F\end{aligned} \qquad (4\text{-}12)$$

T can be calculated by adding the adjusted RT_{NDT}:

T = 42.5 + 104 = 146.5°F　　　　for P = 1105 psig at 54 EFPY

Girth Weld Calculation:

The value of M_m for an inside circumferential postulated surface flaw from Paragraph G-2214.1 [6] was based on a thickness of 6.125 inches (the minimum thickness without cladding); hence, $t^{1/2}$ = 2.47. The resulting value obtained was:

$$M_m = 0.89 \text{ for } \sqrt{t} \leq 2$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464 = \mathbf{1.10}$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

The stress intensity factor for the pressure stress is $K_{Im} = M_m \cdot \sigma$ The stress intensity factor for the thermal stress, K_{It}, is calculated as described in Section 4.3.2.2.4 except that the value of "G" is 20°F/hr instead of 100°F/hr.

Equation 4-9 can be rearranged, and 1.5 K_{Im} substituted for K_{IC}, to solve for (T - RT_{NDT}). Using the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [17], K_{Im} = 25.4, and K_{It} = 2.28 for a 20°F/hr coolant heatup/cooldown rate with a vessel thickness, t, that includes cladding:

$$\begin{aligned}(T - RT_{NDT}) &= \ln[(1.5 \cdot K_{Im} + K_{It} - 33.2) / 20.734] / 0.02 \\ &= \ln[(1.5 \cdot 25.4 + 2.28 - 33.2) / 20.734] / 0.02 \\ &= -53°F\end{aligned} \qquad (4\text{-}12)$$

T can be calculated by adding the adjusted RT_{NDT}:

T = -53 + 104 = 51°F　　　　for P = 1105 psig at 54 EFPY

As stated above, based on the applied pressure and temperature stress intensity factors, the axial weld flaw bounds the P-T curve in the beltline region for 54 EFPY.

Dresden 2

Lower-Intermediate Plate and Vertical Welds

Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 5.7E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2

Lower Plate and Vertical Welds and Girth Weld

Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 4.2E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	54 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower												
6-198-2	A-9128-2	0.20	0.55	143	10	2.9E+17	31	0	15	31	61	71
6-198-3	B-3990-2	0.18	0.51	125	12	2.9E+17	27	0	13	27	54	66
6-198-1	A-9128-1	0.20	0.55	143	30	2.9E+17	31	0	15	31	61	91
Lower-Intermediate												
6-198-12	B4065-1	0.23	0.55	160	20	3.9E+17	41	0	17	34	75	95
6-198-13	B5764-1	0.10	0.50	65	10	3.9E+17	17	0	8	17	33	43
6-198-11	B4030-1	0.20	0.59	148	6	3.9E+17	38	0	17	34	72	78
6-198-9	B4030-2	0.20	0.58	147	-2	3.9E+17	38	0	17	34	72	70
WELDS:												
Lower-Intermediate												
ES*		0.24	0.37	141	23	3.9E+17	36	13	18	45	81	104
SAW**	1P0661/8304	0.17	0.64	158	-5	3.9E+17	41	20	20	57	97	92
	1P0815/8350	0.17	0.52	138	-5	3.9E+17	35	20	18	53	89	84
Lower												
ES*		0.24	0.37	141	23	2.9E+17	30	13	15	40	70	93
SAW**	1P0815/8304	0.17	0.52	138	-5	2.9E+17	30	20	15	50	79	74
Girth												
Lower to Lower-Intermediate												
SAW***	71249/8504	0.23	0.59	168	10	2.9E+17	36	0	18	36	72	82

* Chemistry values are based on data from BAW-2258, dated January 1996, but adjusted. Values of Initial RTndt and σI are obtained from the same document.

** Chemistry values are based on data from BAW-2325, dated May 1998 and Initial RTndt and σI are obtained from the BAW-1803-1, dated May 1991.

*** Chemistry values are based on data from BAW-2325, dated May 1998 and Initial RTndt and σI are obtained from EPRI NP-373, dated 1977.

Dresden 3

Lower-Intermediate Plate and Vertical Welds

Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 5.7E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2

Lower Plate and Vertical Welds and Girth Weld

Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 4.1E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	54 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower												
6-111-2	C1256-2	0.11	0.50	73	-10	2.9E+17	16	0	8	16	31	21
6-111-6	B5159-2	0.24	0.47	153	0	2.9E+17	33	0	16	33	65	65
6-111-7	C1182-2	0.22	0.50	148	10	2.9E+17	32	0	16	32	63	73
Lower-Intermediate												
6-111-3	A0237-1	0.23	0.49	151	10	3.9E+17	39	0	17	34	73	83
6-111-10	B5118-1	0.22	0.49	146	10	3.9E+17	37	0	17	34	71	81
6-111-11	C1290-2	0.15	0.49	104	10	3.9E+17	27	0	13	27	53	63
WELDS:												
Lower-Intermediate												
ES*		0.24	0.37	141	23	3.9E+17	36	13	18	45	81	103.80
Lower												
ES*		0.24	0.37	141	23	2.9E+17	30	13	15	40	70	93
Girth												
Lower to Lower-Intermediate SAW**	299L44/8650	0.34	0.68	221	-5	2.9E+17	47	20	24	62	109	104.16

* Chemistry values are based on data from BAW-2258, dated January 1996, but adjusted. Values of Initial RTndt and σI are obtained from the same document.

** Chemistry values are based on data from BAW-2325, dated May 1998 and Initial RTndt and σI are obtained from the BAW-1803-1, dated May 1991.

Quad Cities 1
Lower-Intermediate Plates and Axial Welds
Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 5.7E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2

Lower Plates and Axial Welds and Lower to Lower-Intermediate Girth Weld
Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 4.1E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	54 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower												
6-122-1	B5524-1	0.27	0.57	180	0	2.9E+17	38	0	17	34	72	72
6-122-2	A0610-1	0.21	0.51	143	-20	2.9E+17	31	0	15	31	61	41
6-122-11	C1485-2	0.23	0.50	153	-10	2.9E+17	33	0	16	33	65	55
Lower-Intermediate												
6-122-4	C1505-2	0.18	0.52	126	-6	3.9E+17	32	0	16	32	65	59
6-122-6	C1498-2	0.17	0.50	119	-20	3.9E+17	30	0	15	30	61	41
6-122-13	A0931-1	0.14	0.51	96	-20	3.9E+17	25	0	12	25	49	29
WELDS:												
Lower-Intermediate												
ES*		0.24	0.37	141	23	3.9E+17	36	13	18	44	81	104
Lower												
ES*		0.24	0.37	141	23	2.9E+17	30	13	15	40	70	93
Girth												
Lower to Lower-Intermediate												
SAW**	72445/8688	0.22	0.54	158	-5	2.9E+17	34	20	17	52	86	81
SAW**	406L44/8688	0.27	0.59	183	-5	2.9E+17	39	20	20	56	95	90

* Chemistry values are based on data from BAW-2259, dated January 1996, but adjusted. Values of Initial RTndt and σI are obtained from the same document.
** Linde 80 weld chemistry values are based on data from BAW-2325, dated May 1998 and Initial RTndt and σI are obtained from the BAW-1803-1, dated May 1991.

Quad Cities 2
Lower-Intermediate Plates and Axial Welds
Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 5.7E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.9E+17 n/cm^2

Lower Plates and Axial Welds and Lower to Lower-Intermediate Girth Weld
Thickness = 6.13 inches

54 EFPY Peak I.D. fluence = 4.1E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 2.9E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	54 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLATES:												
Lower												
6-122-8	C1516-2	0.16	0.46	108	6	2.9E+17	23	0	12	23	46	52
6-122-10	C1501-2	0.18	0.49	124	-10	2.9E+17	26	0	13	26	53	43
6-122-14	C1722-2	0.14	0.54	97	10	2.9E+17	21	0	10	21	41	51
Lower-Intermediate												
6-139-16	C2753-2	0.08	0.50	51	10	3.9E+17	13	0	7	13	26	36
6-139-22	C2868-1	0.08	0.48	51	10	3.9E+17	13	0	7	13	26	36
6-139-25	C3307-2	0.12	0.55	82	10	3.9E+17	21	0	11	21	42	52
WELDS:												
Lower-Intermediate												
ES*		0.24	0.37	141	23	3.9E+17	36	13	18	45	81	104
Lower												
ES*		0.24	0.37	141	23	2.9E+17	30	13	15	40	70	93
Girth												
Lower to Lower-Intermediate SAW	S3986/3870 Linde 124	0.05	0.96	68	-32	2.9E+17	15	0	7	15	29	-3

* Chemistry values are based on data from BAW-2259, dated January 1996, but adjusted. Values of Initial RTndt and σI are obtained from the same document.

RAI 4.2.3

(a) In LRA Section 4.2.3, the applicant states that the original D/QCNPS reflood thermal shock analysis has been superseded by an analysis for BWR-6 vessels that is applicable to the D/QCNPS BWR-3 reactor vessels. Explain why the BWR-6 analysis is applicable to BWR-3 reactor vessel at Dresden/Quad Cities.

(b) In the thermal shock analysis of D/QCNPS reactor vessel core shrouds, presented in LRA Section 4.2.3, the applicant considers the location on the inside surface of the core shroud opposite to the midpoint of the fuel centerline as a location most susceptible to damage during an LPCI thermal shock transient because it receives the maximum irradiation: the 54-EFPY fluence at this location is 5.85×10^{20} n/cm^2 (greater than 1 MeV). This fluence is calculated using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," which is approved by NRC. Confirm whether the effect of extended power uprates, which has been implemented at D/QCNPS, are accounted for in the calculation of the 54-EFPY fluence.

Response:

(a) The BWR/6 evaluation determined the maximum stress intensity in the vessel wall as a function of vessel wall thickness and time after the DBA. As shown in Figure G2214-1 of ASME Section XI, Appendix G (1998 Edition through 2000 Addenda), the stress intensity is a function of vessel wall thickness. The original analysis used a recirculation line break, while the BWR/6 analysis was based on a main steam line break event, which is considered to bound the recirculation line break. In addition, the analysis used a vessel thickness similar to Dresden and Quad Cities vessels. Therefore, the BWR/6 analysis is applicable to the Dresden and Quad Cities reactor vessels.

(b) The fluence used to determine the 54 EFPY shroud fluence was calculated using extended power uprate conditions.

<u>RAI 4.2.4</u>

(a) In LRA Section 4.2.4, "Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and repair Hardware," the applicant calculates the maximum thermal shock stress and the corresponding thermal strain at the location on the inside surface of the shroud receiving the maximum irradiation. The applicant considers this location most susceptible to damage during an LPCI thermal shock transient. The reflood thermal shock would produce high tensile stresses of a short duration on the outside surface of the core shroud, and these stresses are likely to penetrate only to a small depth into the shroud wall. So it appears that the location on the outside surface of the core shroud could be the location most susceptible to damage during an LPCI thermal shock transient.

- Provide an evaluation of strain at the outside surface of the core shroud, exposed to 54-EFPY fluence, during an LPCI thermal shock transient.

- What is the impact of strain rate associated with the LPCI thermal shock transient on the measured and calculated strains in the core shroud?

- The applicant compares the calculated strain range with the measured values of percent reduction in area for annealed Type 304 stainless steel irradiated to 1×10^{21} nvt (greater than 1 MeV) and concludes that the analysis results represent considerable margin of safety. However, it is believed that the calculated thermal strain should be compared with the measured values of percent uniform elongation and not with percent reduction in area. Provide the information about the margin of safety for core shroud in the reflood thermal shock analysis if the calculated strains at both inside and outside surface of the shroud are compared with the measured value of percent uniform strain for annealed Type 304 stainless steel irradiated to 1×10^{21} nvt (greater than 1 MeV).

(b) In LRA Section 4.2.4, the applicant compares the calculated thermal strain with measured values of percent reduction in area for annealed Type 304 SS irradiated to 1×10^{21} n/cm^2. Submit technical basis for comparing the calculated thermal strain with measured values of percent reduction in areas and not with percent uniform elongation.

(c) In LRA Section 4.2.4, the applicant states that the maximum 54-EFPY fluence at the inside surface of the core shroud is 5.85×10^{20} n/cm^2. Since this fluence is greater than the IASCC threshold fluence (5×10^{20} n/cm^2), evaluate the projected accumulated neutron fluence as a TLAA issue for D/QCNPS core shrouds.

Response:

(a)(1) Thermal shock strain is calculated as $\alpha\Delta T/(1-\nu)$

α = coefficient of thermal expansion (9.11x10^{-6} in/in)
ν= Poisson's Ratio (0.3)
ΔT= 540-120 = 420°F (Based on the LPCI transient described in the original analysis, 120°F injection onto a 540°F shroud)
Thermal Strain= (9.11x10^{-6} in/in)(420)/(1-0.3)
Thermal Strain= 0.55%

It should be noted that the original calculation details were not available. However, the above calculations are consistent with the original calculation results.

(a)(2) The thermal strains in the core shroud are calculated based on a linear elastic thermal stress analysis, which is a bounding calculation. The heat transfer coefficient is assumed to be infinite (making the calculation independent of strain rate), and therefore the outside surface of the shroud is considered to be at the fluid temperature (120°F).

At the fluence levels experienced by the shroud, the material will continue to exhibit ductile behavior. The effect of the thermal shock transient is very localized, and the majority of the material is at the higher temperature where the ductility is sufficient to prevent brittle fracture. Even assuming the strain rate has a significant effect, the increased strain rate is still not sufficient to result in brittle fracture. The effect of strain rate during the LPCI thermal shock event can be accounted for by assuming the material yield strength is increased (an effect also produced by increased fluence). At fluence levels up to 1 X10^{21} n/cm^2, (which would represent the increased yield strength), Type 304 exhibits sufficient ductility to preclude brittle fracture.

(a)(3) The strain associated with the reflood thermal shock event is very localized and is constrained by the surrounding bulk material. As such, it is similar to the triaxial stress condition present in the neck region (where the area reduction is taking place) during a tensile test. The percent reduction in area is a measure of this triaxial stress state and, as such, is the most appropriate property for evaluating the effect of thermal shock on the shroud. Therefore, a comparison with uniform elongation is not appropriate in this case.

(b) See response to (a)(3).

(c) See the response to RAI 3.1-7.

<u>RAI 4.2.6</u>

In LRA Section 4.2.6, the applicant states that the procedures and training used to limit cold overpressure events will be the same as those approved by the NRC when Dresden requested to use the BWRVIP-05 technical alternative for the current license term, but it does not explicitly cite a document that support this statement. Provide specific reference(s) in LRA and the UFSAR Supplement that includes the applicant's request to use the BWRVIP-05 technical alternative for the current license term and the NRC approval of that request.

Response:

The procedure and training requirements identified in the Dresden request to use the BWRVIP-05, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, technical alternative were identified in the document listed below:

> Dresden Letter JMHLTR 99-0078 from J. M. Heffley (ComEd) to USNRC, Relief Request for Alternative Weld Examination of Circumferential Reactor Pressure Vessel Shell Welds, July 26, 1999. Attached to Dresden ISI Relief Request No. CR-18.

The NRC approval of the above-listed request and associated procedure and training requirements was provided in the document listed below.

> Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Alternative to Inspection of Reactor Pressure Vessel Circumferential Welds, Dresden Power Station, Units 2 and 3 (attached to USNRC Letter from Anthony J. Mendiola to Oliver D. Kingsley (ComEd), Dresden – Authorization for Proposed Alternative Reactor Pressure Vessel Circumferential Weld Examinations (TAC Nos. MA6228 and MA6229), dated February 25, 2000).

LRA Section 4.2.6 and associated UFSAR Supplement Section A.3.1.6, Reactor Vessel Circumferential Weld Examination Relief, should have referenced the request letter identified above.

RAI 4.2.7

In LRA Section 4.2.7, the applicant calculates the conditional probability of Dresden vessel failure by taking into account the actual inspection of less than 90% of the axial welds instead of essentially 100% of the welds assumed in the calculations by the NRC staff and BWRVIP in support of the elimination of the inspection of the circumferential welds. The analysis concluded that the conditional probabilities of failure due to a low temperature over- pressurization event are very small, 3.89×10^{-8} and 5.07×10^{-8} on a per year basis for Dresden Units 2 and 3, respectively. 10 CFR 50.55.a (g)(6)(ii)(A)(2) states essentially 100% as used in Table IWB-2500-1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component, or part geometry. Was this analysis performed as part of relief from 100% axial and/or elimination of circumferential inspection? What is the impact of 54 EFPY of operation on the probability of vessel failure?

Response:

This analysis was performed to demonstrate that the reliability of the Dresden reactor pressure vessels remained extremely high considering actual inspection coverage. The actual inspection coverage could not meet the "essentially 100%" coverage due to inspection limitations caused by obstructions with internal components and attachments. The analysis did not include the circumferential welds since they had been previously eliminated from the inspection plan in accordance with BWRVIP-05.

The failure probabilities quoted in the question were determined using the predicted fluence at the end of 60 years of operation.

<u>RAI 4.2-BWRVIPS</u>

The NRC staff has approved the applicable BWRVIP reports and attached the following required license renewal applicant action items, in accordance with 10CFR Part 54, when incorporating the reports in a license renewal application:

The license renewal applicant is to verify that its plant is bounded by the report. Further, the renewal applicant is to commit to programs described as necessary in the BWRVIP reports to manage the effects of aging during the period of extended operation. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within these BWRVIP reports described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

10CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA's for the period of extended operation. Those applicants for license renewal referencing the applicable BWRVIP report shall ensure that the programs and activities specified as necessary in the applicable BWRVIP reports are summarily described in the FSAR supplement.

10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. The applicable BWRVIP reports may state that there are no generic changes or additions to technical specifications associated with the report as a result of its aging management review and that the applicant will provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing the applicable BWRVIP reports shall ensure that the inspection strategy described in the reports does not conflict with or result in any changes to their technical specifications. If technical specifications do result, then the applicant must ensure that those changes are included in its application for license renewal.

If required by the applicable BWRVIP report, the applicant referencing a particular report for licensing renewal should identify and evaluate any potential TLAA issues and/or commitments to perform future inspections when inspection tooling is made available.

Provide the necessary commitments, information and changes as described above for each of the following applicable BWRVIP reports:

BWRVIP-74
BWRVIP-05
BWRVIP-38
BWRVIP-76
BWRVIP-75
BWRVIP-25
BWRVIP-27
BWRVIP-48

BWRVIP-18
BWRVIP-26
BWRVIP-41
BWRVIP-47
BWRVIP-49
BWRVIP-78
BWRVIP-86
BWRVIP-42
Other reports applicable to license renewal

Response:

Exelon has reviewed this RAI and has summarized the NRC's information requests and Exelon's response to each of these into the seven elements below.

1. Verify that Dresden and Quad Cities are bounded by the conditions (materials configuration and inspection methodologies) specified in the applicable BWRVIP documents.

   The BWRVIP documents were assembled with participation from the NSSS supplier and a wide representation from the BWR Owners Group, providing a level of confidence in accuracy and bounding conditions of these documents. However, during a preliminary review when preparing this response, some material differences were noted. Exelon will perform a detailed review of the applicable BWRVIP documents and verify that Dresden and Quad Cities are bounded by the conditions specified or identify and evaluate any exceptions noted.

2. Provide a commitment to implement programs consistent with the applicable BWRVIP documents or identify the applicable exceptions.

   At the completion of the review noted in item 1 above, Exelon will provide a list of commitments to the applicable BWRVIP documents or identify specific exceptions taken.

3. Describe how the commitments will be tracked.

   The commitments, once identified will be placed in the site implementing procedures with traceability back to the license renewal commitment being made.

4. Summarize a program description of the applicable BWRVIP documents in the LRA Appendix A, UFSAR Supplement.

   Several of the BWRVIP programs are identified in the LRA Appendix A, such as BWRVIP-75, A.1.7; BWRVIP-27, A.1.8; BWRVIP-48, A.1.4; BWRVIP-49, A.1.8; BWRVIP-78, A.1.22; and BWRVIP-86, A.1.22. Once the comprehensive list of commitments is identified in item 2 above, Exelon will update the LRA Appendix A to provide a summary program description to address each applicable BWRVIP document.

5. Verify technical specification changes needed to support implementation of the applicable BWRVIP documents have been identified and processed.

   There are no additional Technical Specification changes anticipated. However, once the detailed review summarized in item 1 above is complete Exelon will confirm no Technical Specification changes are needed or identify the needed changes to be processed prior to the start of extended term of operation.

6. Identify and evaluate any potential TLAA issue identified by the applicable BWRVIP documents.

All applicable TLAA's are discussed in Section 4 of the LRA

7. Address items 1 through 6 above for the 16 specific BWRVIP documents listed in the RAI and identify and address other BWRVIP documents applicable to license renewal.

Based on a preliminary review there appears to be several other BWRVIP documents applicable to license renewal, such as BWRVIP-07 and 63 for Core Shroud Repairs, and BWRVIP-26 for Water Chemistry. Once the detailed review is completed Exelon will provide an amended response addressing items 1 through 6 for all BWRVIP documents applicable to license renewal.

RAI 4.2-FLAW EVALUATION

Have there been any flaws that were left in service based on ASME Code Section XI analysis techniques? If so, did you consider such analyses as potential TLAA's?

Response:

Flaws have been left in service based on ASME Code Section XI analysis techniques. The analyses associated with such flaws were reviewed and considered as potential TLAAs. None of these flaw analyses were determined to be TLAAs because the analyses did not satisfy Criterion (3) of 10 CFR 54.3, Definitions, Time Limited Aging Analyses. The analyses did not involve time-limited assumptions defined by the current operating term.

RAI 4.7.2.3

(a) Describe the bases for the TLAA calculation in sufficient detail to justify the "assumed" corrosion loss of 4 mils/year for 33 years. State the nature of this design calculation including what was calculated and how the corrosion loss of 4 mils/year was included in the calculations. Substantiate that this is a bounding corrosion rate for all foreseeable conditions in both reactors. For example, there are instances of release of ion exchange resins into BWR reactor coolant systems. Are credible nonstandard water chemistry conditions accounted for in the calculation?

(b) A single ultrasonic inspection is proposed to confirm the assumptions used in the corrosion rate calculations for galvanic corrosion in the Containment Shell and Attached Piping Components. Describe how the location for the single bounding ultrasonic inspection will be selected to assure that it represents the most aggressive corrosion conditions for both sites.

(c) State the corrective measures that will be taken in the event that the revised galvanic corrosion calculation indicates an unacceptable wall thickness prior to the end of the 60-year licensed operating period.

Response:

(a) The subject calculation was performed to evaluate and qualify the bolted flange connections between the ECCS suction strainers and associated torus penetration nozzles. The calculations were performed in response to an NRC concern identified during a review of the modification associated with replacement of the ECCS suction strainers (NRC Inspection Report 50-237/01-09; 50-249/01-09). The NRC believed that the original calculation did not sufficiently account for the affects of galvanic corrosion. Each of the bolted flange connections consists of an existing carbon steel flange welded to the associated torus nozzle and a new stainless steel slip-on flange welded to the strainer body. The stated corrosion allowance contained was used to determine the radial stresses at the bolt circles of the existing carbon steel flanges in accordance with ASME Section III, Subsection NC, 1977 including Summer Addenda. The corrosion allowance used in the revised calculation was conservatively assumed to occur on the entire perimeter of all bolt circles simultaneously. The calculated stresses were then used to calculate the maximum interaction ratios. A maximum interaction ratio < 1.0 indicates that stresses are within limits.

The basis for the corrosion rate used in the revised calculation (4 mils/year) was obtained from Uhlig's Corrosion Handbook, 2nd Edition. The revised calculation did not provide any specific consideration of nonstandard water chemistry conditions. However, the subject strainers are located in the suppression chambers of the respective units. Water quality in the suppression chamber is maintained within strict limits and periodically (quarterly) sampled as governed by corporate and station procedures. Water quality sampling is frequent enough to allow prompt identification and correction of any nonstandard water chemistry conditions capable of increasing corrosion of the associated flanges. Use of the subject corrosion rate is discussed in NRC inspection report 50-237/01-09; 50-249/01-09.

(b) The location of the UT inspections will be randomly selected. Initial thickness measurements will be made at 2 to 4 adjacent bolt locations on one flange. Separate thickness measurements of the same locations will be made in a subsequent outage, thereby establishing the actual corrosion rate. These inspections will be performed on one flange per unit at Dresden Station. This method of selection is acceptable to assure that results are representative of the most aggressive corrosion conditions at both sites because:

- Water chemistries are similar at each site and required to be maintained within limits established by procedures.
- The strainers at Dresden and Quad Cities were installed in approximately the same timeframe (1997/1998).
- The strainer/flange configurations are similar at each site.

(c) In the event that the revised galvanic corrosion calculation indicates that an unacceptable wall thickness will be reached prior to the end of the 60-year licensed operating period, the Exelon Corrective Action Program will be used to develop appropriate corrective actions. The following may be taken as part of the corrective actions, but the exact corrective actions cannot be determined until the actual conditions are determined:

- Inspect additional flanges.
- Establish root causes and corrective actions including the possible creation of periodic UT inspections of the affected flanges.
- Investigate possible modification and/or replacement of the affected flanges and/or interfacing components.

RAI 4.7.3

(a) Section 4.7.3 of the LRA states that flaw evaluations were performed as a TLAA for Dresden and Quad Cities to evaluate the potential effects of arc strikes. The discussion makes reference to both postulated flaws and to flaws that were repaired. Clarify if flaws were actually detected at the arc strike locations, if any repairs of such flaws were made, and/or if the flaws of concern were only postulated for purposes of fracture mechanics evaluations of structural integrity.

(b) Section 4.7.3 of the LRA cites crack growth evaluations that were performed to address the effects of arc strikes on the wall of the suppression chamber. Were fracture mechanics methods and acceptance criteria of ASME Section XI used for these evaluations? If not, describe the alternative methods and acceptance criteria that were used.

Response:

(a) Arc strikes discovered in the Quad Cities Unit 2 suppression pool shell were repaired by grinding and blending the flaws in the shell. A crack growth analysis was performed to support continued operation assuming a postulated defect in the heat affected zone. This analysis was determined to be a TLAA because the calculation assumed an operating limit of 850 SRV load cycles to justify continued operation without any further repair. A UT measurement performed in 1993 validated that there was no flaw in the heat affected zone and no further repairs to heat affected material were performed.

An analysis was completed in October 1991 at Dresden Station to evaluate suppression chamber pitting following repairs (grinding and blending) to two corrosion pits discovered in Unit 3. The crack growth analysis applied to the Quad Cities Unit 2 arc strikes was used to bound the Dresden corrosion pitting as the postulated flaw size at Quad Cities was more restrictive than the geometry existing at Dresden. This analysis concluded additional repairs or modifications at Dresden were not required.

In 1994, a generic crack growth analysis was completed to disposition torus shell anomalies caused by heating processes and non-heating processes. This analysis also assumed the same operating limit of 850 SRV cycles to justify continued operation without repair provided the flaw depth was no greater than 0.0625 inches. This analysis was determined to be a TLAA because the calculation assumed an operating limit of 850 SRV load cycles to justify continued operation. This same analysis has been used to justify additional corrosion pitting flaws discovered at both sites since 1994 without repair.

(b) Both analysis described in the response above involved fracture mechanics evaluation that applied the acceptance criteria of ASME Code, Section III as defined in NUREG-0661 (Safety Evaluation Report, Mark 1 Containment Long Term Program, Resolution of Generic Technical Activity).

RAI 4.7.4

In Section 4.7.4 of the LRA, a test is referenced that was conducted to establish the threshold for loss of resilience of the polyurethane used in the expansion gap. In conducting the test, what factors (e.g., temperature and environmental, such as moisture and/or oxygen at maximum credible levels, factored into the test) were considered to have an important influence on the radiation stability of the polyurethane? What other factors were considered, but were judged to be relatively unimportant? Were the important factors covered by the test conditions? Justify that the test results comprise a satisfactory basis for predicting the acceptable dose to within required uncertainty limits.

(RAI 3.5-10 also refers to TLAA Section 4.7.4)

Response:

The drywell shell is largely enclosed within the concrete of the primary containment shield wall. To accommodate both thermal and pressure caused expansion, an expansion gap was provided between the concrete and the drywell shell. The polyurethane foam in the expansion gap was a construction aid used to permit pouring the concrete support structure over the steel drywell shell while maintaining the required expansion gap. The expansion gap foam remains in place over the lifetime of the plant. During thermal or pressure transients that cause expansion of the steel drywell shell, the expansion gap foam exerts forces on the drywell shell in opposition to the shell's expansion. To ensure that forces exerted by the expansion gap foam remain within drywell shell design limits over the lifetime of the plant, the effect of radiation on the resilient characteristics of polyurethane was considered. The UFSAR states that the polyurethane foam will be exposed to a maximum of 2.5 X 10^7 rads based on 40 full years of reactor operation (assuming operation at the initial licensed operating power level) and that there is no detectable change in resilience below 10^8 rads. LRA Section 4.7.4 evaluates the resilient characteristics of the polyurethane foam as a TLAA by determining the maximum exposure based on 60 years operation at approved increases in power rating. The maximum exposure is determined to be 4.2 X 10^7 rad. Because this exposure is less than 10^8 rad, the resilient characteristics of the polyurethane foam are validated as acceptable for the 60-year extended operating period based on the data contained in the UFSAR.

The reference to a test in LR Section 4.7.4 is based on the following text contained in both the Dresden UFSAR and the Quad Cities UFSAR, Section 6.2.1.2.1.1 (Drywell Expansion Gap):

> Polyurethane foam samples, similar to that used in the gap, were irradiated at various levels from 10^7 to 10^9 rads. There was no detectable change in resilience below 10^8 rads, thus amply confirming the published data.

The same wording was contained in Section 5.2.3.7 of the original Quad Cities FSAR. As such, the description of irradiation testing of polyurethane foam samples is more than 30 years old. Exelon has no additional details about the test described in the UFSAR.

RAI B.1.1

(a) The LRA states that the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is part of the inservice inspection (ISI) program and provides for condition monitoring of reactor coolant pressure retaining piping and components within the scope of license renewal. The LRA goes on to state that the program includes crack monitoring for susceptible inservice inspection Class 1 components subject to a steam or reactor water environment, through volumetric examinations of pressure retaining welds and their heat affected zones in piping components. Is the intent to limit volumetric inspections only to Class 1 piping? Please explain.

(b) The LRA states that loss of fracture toughness monitoring for susceptible inservice inspection Class 1 components in reactor recirculation and reactor water cleanup systems will be accomplished by performing visual examinations of Class 1 reactor recirculation and reactor water cleanup system valves and reactor recirculation pumps. NUREG-1801 XI.M12 concluded that all the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies. Since these Code requirements include volumetric/surface exams of welds and visual examinations (VT-3 on the internal surfaces and VT-2 pressure tests), this AMP should therefore identify what particular Code inspection activities are applicable for Dresden and Quad Cities and if the alternative ASME Code Case N-481 will be invoked.

(c) The LRA states that surface and volumetric examinations will be performed to monitor cracking in reactor internal components subject to a reactor water environment. Is this the same thermal aging management activity described in B.1.9? If so, B.1.1 should be changed to indicate that reactor vessel internal VT-1 and VT-3 examination requirements of ASME Code, Section XI, Table IWB 2500-1 shall be augmented according to AMP B.1.9. Since NUREG-1801 allows the use of the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry, the LRA should state in B.1.1 or B1.9 whether this alternative is applicable to Dresden and/or Quad Cities.

(d) By letters dated September 5, 2001 (ADAMS #ML012050103) and February 5, 2002 (ADAMS #ML020180003) alternative risk-informed inservice inspection (RI-ISI) programs were approved for Dresden Nuclear Power Station Units 2 & 3 (Dresden) and for Quad Cities Nuclear Power Station, Units 1 & 2, (Quad Cities) for ASME Class 1 and 2 piping. These programs are an alternative to the ASME Section XI program and examination requirements for category B-J, B-F, C-F-1, and C-F-2 piping components and result in significant differences in the locations examined, the scope of examinations and the type of examinations performed when compared to the requirements specified in ASME Section XI Subsections IWB and IWC for piping. However, the RI-ISI alternative is not specifically addressed in NUREG-1801 (GALL) aging management program XI.M1. Therefore, clarify as to whether ASME Section XI Subsection IWB and IWC program requirements or an alternative RI-ISI program will be used for Class 1 and 2 piping within the scope of license renewal. If the alternative RI-ISI program will be used, revise the AMP B.1.1 accordingly. Has the plant specific RI-ISI evaluations identified any particular risk significant components subject to aging management or particular aging effects (degradation mechanisms) not addressed in the GALL report?

(e) B.1.1 does not make reference to augmented inspections of Class 1 piping < 4-inch NPS discussed in B.1.23. As stated in RAI B.1.1(d) above, the examination requirements under the alternative RI-ISI program are significantly different. The examination requirements for the Dresden and Quad Cities RI-ISI programs are consistent with EPRI TR-112657 and where appropriate require volumetric examinations of piping < 4-inch NPS. Since it is expected that Dresden and Quad Cities will implement RI-ISI, will credit be taken for volumetric inspections performed on small bore Class 1 piping < 4-inch NPS performed as part of the RI-ISI alternative?

(f) For the isolation condenser, stress corrosion cracking (SCC) and cyclic loading are the crack aging mechanisms identified in LRA Table 3.1-1; however, in B.1.1 the LRA states that crack monitoring of the Dresden isolation condenser includes surface and volumetric examinations of the pressure retaining nozzle welds and their heat affected zones. The LRA states that crack monitoring of the Dresden isolation condenser is performed through surface and volumetric examinations of pressure retaining nozzle welds and heat affected zones that are subject to a steam or reactor water environment. Please describe examinations that will be performed to identify the presence of SCC in the isolation condenser stainless steel tubing.

(g) FSAR Supplement A.1.1 needs to be revised to make reference to NUREG-1801 XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" and the implementation of RI-ISI if applicable.

Response:

(a) The intent was not to limit volumetric examinations to Class 1 components. The program description included in Section B.1.1 of Appendix B provides a general summary of the major components monitored by this program. Crack monitoring of the Reactor Internals and the Isolation Condenser are examples where non-Class 1 components will receive volumetric exams.

(b) NUREG-1801, Line IV.C1.1-g, which references aging management program XI.M12, does not apply to either Dresden or Quad Cities as neither plant has CASS piping. As such, NUREG-1801 aging management program XI.M12 was not credited.

The examinations for the reactor recirculation and reactor water cleanup system valves and reactor recirculation pumps as identified in the LRA are consistent with NUREG-1801, Lines IV.C1.2-c, and IV.C1.3-b. These NUREG-1801 lines recommend Aging Management Program XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD". These lines also state, "For pump casings (and valve bodies), screening for susceptibility to thermal aging is not required. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS valve bodies." Dresden and Quad Cities are implementing the requirements of ASME Section XI, Table IWB-2500-1, Item B12.20 and B12.50, which require a VT-3 of the internals of these components. Code Case N-481 has not used. Therefore no additional inspections or evaluations are required.

(c) The components subject to the examinations in aging management program B.1.1 are the access hole cover plates and attachment welds as specified in NUREG-1801, Lines IV.B1.1-d and IV.B1.1-e. The components subject to examination by aging management program B.1.9 are Core Shroud (NUREG-1801, Line IV.B1.1-a), Core Plate (NUREG-1801, Line IV.B1.1-b), Core Shroud and Core Plate Shroud Support Structure (NUREG-1801, Line IV.B1.1-f), Top Guide (NUREG-1801, Line IV.B1.2-a), Core Spray Lines and Spargers (NUREG-1801, Line IV.B1.3-a), Jet Pump Assemblies (NUREG-1801, Line IV.B1.4-a), Fuel Supports and Control rod Drive Assemblies (NUREG-1801, Line IV.B1.5-c), and Instrumentation (NUREG-1801, Line IV.B1.6-a). Exelon has not submitted relief based on BWRVIP-62 "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection" to the NRC for Dresden or Quad Cities. As such, BWRVIP-62 was not credited for aging management.

(d) LRA Appendix B, B.1.1 should have noted an exception for the implementation of Risk Informed Inservice Inspection (RI-ISI). Both Dresden and Quad Cities will be implementing RI-ISI and its alternative inspections for Class 1 and 2 piping within the scope of license renewal. The plant specific RI-ISI evaluations have not identified any particular risk significant components subject to aging management or particular aging effects (degradation mechanisms) not addressed in NUREG 1801.

(e) As discussed in the response to RAI 3.1-9, Class 1 piping <4-inch NPS will be inspected based on based on RI-ISI. See RAI 3.1-9 for further details.

(f) LRA Table 3.1-1, Ref. No. 3.1.1.7, refers to exception and further evaluation paragraphs (3.1.1.2.3 and 3.1.1.1.7 respectively). Both of these paragraphs indicate that the B.1.1 program will be augmented by the B.2.6 "Heat Exchanger Test and Inspection Activities" program. The augmented inspections include eddy current testing of the stainless steel tubes to detect stress corrosion cracking.

(g) The aging management programs described in Appendix B of the LRA are not always consistent with equivalent programs described in NUREG-1801. To avoid any confusion, Exelon made a decision to not use the NUREG-1801 aging program numbering designations to describe the programs contained in Appendix B. However, Table B-1, NUREG-1801 and Dresden and Quad Cities Aging Management Program Matrix, does provide a list of aging management programs evaluated in Chapters X and XI of NUREG-1801 and provides the corresponding Dresden and Quad Cities programs credited for aging management. For this reason, the FSAR Supplement (Appendix A) references the program titles as described in Appendix B. The ISI Program in Appendix A reflects a summary of the LRA Appendix B Program with its enhancements as opposed to the corresponding NUREG-1801 program.

However, as noted in item (d) above the Appendix B program B.1.1 and Appendix A, A.1.1 should have noted the use of RI-ISI as an exception to NUREG-1801.

RAI B.1.2

(a) The GALL report references EPRI TR 103515 Revision 1 for guidance on Water Chemistry programs, whereas the applicant's Water Chemistry program references Revision 2 of the EPRI guidance. Outline key differences in the two revisions and justify why Revision 2 is acceptable for application to Dresden and Quad Cities.

(b) The applicant's Water Chemistry AMP is guided by EPRI TR-103515, the 2000 revision of "Water Chemistry Guidelines for Power Operation." The applicant indicates that hydrogen peroxide is not measured because rapid decomposition makes measurements exceptionally difficult to obtain, and the EPRI Guidelines do not address monitoring for hydrogen peroxide. As hydrogen peroxide is decomposed, are there locations in any system covered by the Water Chemistry AMP where radiation is sufficient to generate additional hydrogen peroxide, resulting in significant steady state concentrations? The GALL report indicates that "hydrogen peroxide is monitored to mitigate degradation of structural materials." Justify that steady state hydrogen peroxide concentrations are below thresholds that prompted the issue raised in GALL or indicate what actions the applicant will take to investigate whether structural degradation in potentially affected locations is ongoing.

(c) The applicant indicates that pH is not monitored "because pH measurement accuracy in most BWR streams is generally suspect because of the dependence of the instrument reading on ionic strength of the sample solution," citing the 2000 revision of the EPRI guidelines. Some phenomena, e.g., flow-accelerated corrosion, have, in the past, been characterized by water chemistry parameters that include pH. In lieu of direct pH measurements, indicate whether alternative methods are applied to characterize the aggressiveness of the water chemistry. If so, describe the method(s) and how they are implemented to assure that water chemistry remains within parameters that will not result in degradation that will jeopardize the safety function of any system or component. Also, the applicant stated that dissolved oxygen is not monitored for certain components/water. Does the Water Chemistry Program monitor dissolved oxygen in reactor water? If not, please explain why monitoring for dissolved oxygen is not necessary.

(d) The Water Chemistry AMP, B.1.2, states that aging management for the SBLC system relies on monitoring of and control of SBLC makeup water chemistry because the sodium pentaborate solution masks chemistry measurements. Thus, conditions in the storage tank are not monitored. Given that there are instances of out of specification chemicals, provide assurance that the receipt inspection process will preclude introduction of unexpected impurities with the sodium pentaborate to avoid aggressive conditions in the tank.

(e) As recommended in Table C-2 of Appendix C, EPRI Report TR-103515, "BWR Chemistry Guidelines," provisions for increased frequencies of the torus water chemistry should be included in the station procedures if chemical ingress is detected or suspected. Confirm that this is done; if not, justify.

(f) UFSAR Supplement A.1.2 needs to be changed to make reference to NUREG-1801 XI.M2, "Water Chemistry" and the implementation of changes to water chemistry control per the 2000 revision of EPRI-TR-103515 "BWR Water Chemistry guidelines."

(g) The One-Time Inspection AMP (B.1.23) includes provisions specified by the GALL report to specifically address areas of low-flow in systems that are covered by the Water Chemistry AMP. Indicate how the One-Time Inspection Program will be applied to the most vulnerable areas, the basis for selection of these areas and how these areas are applicable to other system locations covered by the Water Chemistry AMP. Will the one time inspection be able to confirm the effectiveness of the AMP to manage aging effects in areas of low flow and for other areas subject to degradation if the management of water chemistry is inadequate? (Refer to RAI B.1.23-1)

(h) The Water Chemistry Program is credited with managing aging in aluminum water storage tanks. This material is not within the scope of the GALL materials, yet it is not identified as an exception in the Water Chemistry Program. Indicate how the Water

Chemistry Program will manage the effects of aging of the aluminum tanks. Describe aging mechanisms for the aluminum that are of concern to water storage tanks and identification of an aging management program.

(i) Describe evidence from operating experience which demonstrates that the existing Water Chemistry AMP has been successful in mitigating aging effects. In particular, the applicant's section on Operating Experience indicates that there have been instances when water chemistry parameters have been outside established specifications and the applicant indicates under "Enhancements" that procedures will be revised for increased sampling frequency to verify corrective actions taken to address abnormal chemistry conditions. Discuss the abnormal chemistry conditions, the resulting scope of increased sampling, and whether there were or will be assessments/inspections of potential impacts on affected system/component materials. If not, justify that assessments/inspections are not needed. What actions were taken to investigate whether the excursions resulted in age-related degradation? Is increased sampling of reactor water chemistry included in the "enhancements?" If so, provide specifics of the increases in sampling. In not, provide justification.

Response:

(a) The key differences between EPRI TR-103515, Revision 1, "BWR Water Chemistry Guidelines" – 1996 Revision and EPRI TR-103515, Revision 2, "BWR Water Chemistry Guidelines" – 2000 Revision are as follows. Additional discussion is provided to describe how each of the key differences has been implemented at Dresden and Quad Cities and provide justification that the change, as implemented, is acceptable.

1. In Revision 2 to the EPRI BWR Water Chemistry Guidelines, chlorides and sulfates no longer need to be measured on a daily basis provided that reactor water conductivity is trended to ensure that the action level 1 limits are not exceeded. Dresden and Quad Cities has not incorporated this change into the plant chemistry procedures. Both sites continue to measure chloride and sulfate levels daily. Plant implementation has not resulted in any change from the guidance provided in EPRI TR 103515, Revision 1.

2. In Revision 2 to the EPRI BWR Water Chemistry Guidelines, plants with hydrogen water chemistry (HWC) or HWC with Noble Metals Chemical Addition (NMCA) no longer need to measure electrochemical corrosion potential (ECP) on a continuous basis. Dresden and Quad Cities utilize HWC with NMCA and use ECP monitoring. Plant implementation has not resulted in any change from the guidance provided in EPRI TR 103515, Revision 1.

3. Revision 2 to the EPRI BWR Water Chemistry Guidelines, allows plants with HWC or HWC with NMCA to go to higher action level 2 and 3 levels for chloride and sulfate. Action level 2 was increased from >20 ppb to > 50 ppb and Action level 3 was increased from >100 ppb to > 200 ppb. This additional flexibility is allowed based on the increased protection of reactor coolant system and reactor assembly components provided by HWC or HWC with NMCA. The increased level values have been incorporated into the Dresden and Quad Cities chemistry procedures. Dresden and Quad Cities chemistry procedures require action to be taken to return these parameters to the desired range if these parameters exceed the goal values, and the goal values are well below the Action Level 2 and 3 values provided in EPRI TR

103515, Revision 1. Plant implementation is acceptable because goal values are less than the action level values provided in EPRI TR 103515, Revision 1.

4. Revision 2 to the EPRI BWR Water Chemistry Guidelines added Reactor Water Iron as a new diagnostic parameter. Dresden and Quad Cities have implemented this change. This is a new diagnostic parameter. Adding a new diagnostic parameter is acceptable because it is a conservative change from the guidelines of EPRI TR 103515, Revision 1.

5. Revision 2 to the EPRI BWR Water Chemistry Guidelines reduced the Action Level 1 limit for feedwater copper from >0.5 ppb to >0.2 ppb. The decreased level 1 value has been incorporated into the Dresden and Quad Cities chemistry procedures. The new value is more conservative then the previous, so this change from the guidelines of EPRI TR 103515, Revision 1 is acceptable.

6. Revision 2 to the EPRI BWR Water Chemistry Guidelines revising the action level 1 limit for dissolved oxygen in the feedwater from a minimum of 15 ppb to a minimum of 30 ppb. Testing concluded that the flow accelerated corrosion (FAC) wear rate is inversely proportional to the concentration of dissolved oxygen therefore this change is conservative. The increased action level 1 limit has been incorporated into the Dresden and Quad Cities chemistry procedures. This is a conservative change from the guidelines provided in EPRI TR 103515, Revision 1.

The provisions of Revision 2 of EPRI TR-103515 were found acceptable by the NRC Staff in NUREG-1769 Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3.

(b) Reliable hydrogen peroxide data is exceptionally difficult to obtain. Decomposition of hydrogen peroxide to water and oxygen in reactor coolant sample lines is very rapid and Exelon has no data with regard to locations where radiation is sufficient to generate additional hydrogen peroxide resulting in significant steady state concentrations. The oxygen level in the reactor water is continuously monitored at Dresden and Quad Cities, thereby offering some indication as to the level of hydrogen peroxide. Computer simulation of water radiolysis can describe concentrations of hydrogen peroxide in the various parts of the BWR primary circuit and in the main steam. Radiolysis modeling predicts that hydrogen peroxide is the major oxidizing constituent formed in the BWR vessel. Hydrogen addition to feedwater has been applied in order to mitigate occurrence of IGSCC of structural materials by suppressing the formation of hydrogen peroxide. The hydrogen addition has accomplished an Electrochemical Corrosion Potential (ECP) value less than -230mV, SHE (Standard Hydrogen Electrode). By maintaining a low ECP less than –230mV, SHE, the reactor water chemistry minimizes the effects from hydrogen peroxide below the threshold that prompted the issue raised in NUREG 1801.

Dresden and Quad Cities uses the ISI program to investigate whether structural degradation in potentially affected locations is ongoing. The Dresden and Quad Cities ISI program provides for condition monitoring of the reactor vessel, reactor internal components and ASME Class 1 pressure retaining components in accordance with ASME Section XI, Subsection IWB. Indications and relevant conditions detected during examinations are evaluated in accordance with ASME Section XI Articles IWB-3000, for Class 1.

(c) Alternate methods are applied to monitor the water chemistry of the condensate storage tank, demineralized water storage tank and the torus (pressure suppression pool) in lieu of direct pH measurements. The Dresden and Quad Cities chemistry procedures require the monitoring of conductivity, chlorides and sulfates in accordance with limits set by EPRI TR-103515.

The Dresden and Quad Cities procedures set goal values, which are below the limit values set by EPRI TR-103515. When a monitored parameter exceeds the goal values, the procedure requires that the values be confirmed, corrective action be taken to return the parameter to the desired range, and that increased sampling be performed to verify the effectiveness of the corrective action to address the abnormal chemistry condition.

In reactor water, dissolved oxygen is sampled in accordance with guidance provided in EPRI TR-103515 and NUREG 1801 XI.M2 Water Chemistry. The exception to Water Chemistry for dissolved oxygen, LRA Table B.1.2 states "condensate storage tank, demineralized water storage tank water and torus (pressure suppression pool) water is not sampled for dissolved oxygen."

(d) Borax and Boric Acid, which are mixed to produce the sodium pentaborate, are purchased in accordance with the requirements listed in GE Material Specification D50YP1, Revision 3, which are contained the Dresden and Quad Cities Standby Liquid Control System Material Specification. The receipt inspection verifies that the vendor complied with those requirements listed in the material specification.

(e) Dresden and Quad Cities torus chemistry procedures state that analysis frequencies other than those specified may be requested by Chemistry Operations Departments. In the event of chemical ingress, whether detected or suspected, the torus water chemistry procedures provide guidance to the chemistry department with regard to increasing the sampling frequency. The procedures require the notifying of chemistry supervision of an out-of-specification value and to initiate corrective actions to return the parameter to desired range. In addition, whenever corrective actions are taken, increased sampling is utilized to verify the effectiveness of the corrective actions.

(f) Exelon followed the examples in Tables 3.1-2, 3.2-2, 3.3-2, 3.4-2 and 3.5-2 (Water chemistry), NUREG 1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants to compose the words contained in A.1.2. NUREG 1800 does not refer to the revision of EPRI TR-103515 and does not refer to the NUREG 1800 XI.M2 program. Section 3.1.3.4 of NUREG 1801 states that the reviewer should verify that the applicant has provided information equivalent to that contained in Table 3.1-2. Exelon believes that current wording of A.1.2 meets the requirements of NUREG 1800 to provide information equivalent to that in NUREG 1800 Tables 3.1-2, 3.2-2, 3.3-2, 3.4-2 and 3.5-2 (Water chemistry). As such, no change to A.1.2 is required.

(g) One-time inspections are scheduled for implementation prior to the period of extended operation to verify the effectiveness of the water chemistry program. The one-time inspections will be conducted using work orders, which provide instructions on the type of inspection, the acceptance criteria, and requirements for evaluation.

The most vulnerable areas selected for one-time inspections were based upon having stagnant flow. These areas are susceptible to general, crevice, and pitting corrosion. In addition, since oxygen is needed to initiate all three types of corrosion, a location that encounters occasional flow was selected so that the oxygen supply can be replenished.

General corrosion is more prevalent in carbon steel, and pitting and crevice corrosion is more prevalent in stainless steel; therefore, an inspection of both types of materials will be performed.

The following are the carbon steel components selected for inspection.

- The HPCI torus suction check valves are carbon steel typically exposed to torus water. The conditions are typically stagnant flow but with occasional flow when the HPCI system is activated. The joints and connections in the valves offer an inspection point for crevice corrosion.

- The HPCI Booster Pump Casing is carbon steel typically exposed condensate storage tank water. The conditions are typically stagnant flow but with occasional flow when the HPCI system is activated. The joints and connections in the pump offer an inspection point for crevice corrosion.

The following are the stainless steel components to be inspected.

- The Control Rod Drive (CRD) system contains stainless steel valves, exposed to condensate storage tank water, which are typically under stagnant flow conditions but occasionally flow exists when the system is activated. The joints and connections in the valves offer an inspection point for crevice corrosion.

- The Dresden Spent Fuel Pool Cooling and Demineralizer System contains stainless steel piping exposed to spent fuel pool water, which are typically under stagnant flow conditions but occasionally flow exists when the system is activated.

- The SBLC System contains stainless steel pumps and valves exposed to a stagnant sodium pentaborate environment. The solution is a mixture of CST makeup water and sodium pentaborate. Therefore the effectiveness of the CST water chemistry program is verified. The joints and connections in the pump and valves offer an inspection point for crevice corrosion.

The above listed components were selected based on the materials of construction and the conditions present that would optimize the initiation of general, pitting, and crevice corrosion due to the typically stagnant flow but with the occasional replenishment of the oxygen. These components are representative of the worst case for all the components that are managed for aging by the Water Chemistry Aging Management Program.

If the results of the one-time inspections indicated that these components are not experiencing degradation, the effectiveness of the water chemistry program will be validated. For test or inspection results that do not satisfy established criteria, an evaluation will be performed and a condition report will be initiated to document the concern in accordance with plant administrative procedures.

(h) NUREG 1801 specifies Aging Management Program characteristics but does not specify materials to which the Aging Management Program applies. Therefore, material that is not within the scope of NUREG 1801 is not classified as an exception to the NUREG 1801 Aging Management Program. The Water Chemistry Program was determined to be an acceptable program to manage the effects of aging of the aluminum tanks.

Aluminum is a reactive metal, but develops a strongly bonded oxide film, which gives it excellent corrosion resistance in most environments. Once damaged, this film reforms immediately. Aluminum can be susceptible to attack by both crevice and pitting corrosion.

Maintaining a low impurity environment can minimize crevice and pitting corrosion. EPRI TR-103515, "BWR Water Chemistry Guidelines," Rev. 2, Table B-1, sets the limits for the concentration of corrosive impurities such as chlorides and sulfates below the levels known to cause loss of material. The Dresden and Quad Cities water chemistry procedures control the level of chlorides and sulfates in accordance with the EPRI Chemistry Guidelines for the storage tanks. Conductivity is also monitored in accordance with EPRI Chemistry Guidelines, which would give an indication of the introduction of impurities into the tanks.

Given the excellent corrosion resistance of aluminum compared to carbon and stainless steel, the Dresden and Quad Cities Water Chemistry Program will adequately manage the aging of the aluminum storage tanks by maintaining low water impurities.

In addition, the Dresden and Quad Cities Buried Piping and Tanks Inspection Program (AMP B.1.25) includes a one-time internal UT of the bottom of the aluminum Condensate Storage Tank or Demineralized Water Storage Tank. The internal UT of the tanks will validate the effectiveness of the Aging Management Program.

(i) A review of operating experience at Dresden or Quad Cities did not note any degradation that could be attributed to abnormal chemistry conditions.

The following water chemistry parameters were observed to be outside established specifications:

<u>Reactor water chemistry - Quad Cities</u>

- Elevated chlorides were detected due to condenser leaks and river in-leakage from the crib house due to high river level.
- Increases of conductivity were noted due to hideout return from calcium, magnesium, and sulfate.
- Elevated feedwater iron concentration occurred due to a feedwater heater trip, perturbation by starting and stopping of feedwater pumps or condensate pumps, and due to a restructuring of the sample line oxide layer when the flow of zinc injection was increased.
- Low dissolved oxygen and high conductivity of feedwater were observed after the hydrogen addition system was placed in service.
- Occurrences of elevated sulfate in reactor water have been reported due to reactor water cleanup system out of service, flow fluctuation through demineralizer bed, and resin bleed through more pleated filters.

<u>Reactor water chemistry - Dresden</u>

- Air in-leakage through the isolation valve of a condensate pump that was taken out of service for preventative maintenance caused elevated feedwater oxygen.
- Occurrences of elevated sulfate in reactor water have been reported due to

reactor water cleanup system out of service, flow fluctuation through demineralizer bed, and resin bleed through more pleated filters.

Condensate Storage Tank (CST) and Demineralized Water Storage Tank (DWST) water chemistry - Quad Cities

- Incidents of elevated conductivity and dissolved oxygen concentration were reported in control rod drive system water chemistry. Causes identified were inadequate condensate reject flow, use of alternate source from the air saturated CST due to condensate reject valve being out of service, and loss of loop seal that introduced air into the condensate.
- An increase in silica concentration occurred with the reactor shutdown as a result of water transfer from the torus to CST.
- Control rod drive system dissolved oxygen level reached Action Level 1 before condensate reject flow was raised.

Condensate Storage Tank (CST) and Demineralized Water Storage Tank (DWST) water chemistry - Dresden

- No excursions identified.

Spent Fuel Pool water chemistry

- No excursions identified for Dresden or Quad Cities.

Suppression Pool / Torus chemistry - Quad Cities

- The suppression pool water conductivity was over the administrative limit for conductivity due to work going on in the suppression pool to replace an instrument air line hanger.

Suppression Pool / Torus chemistry – Dresden

- Water samples taken on the low pressure coolant injection (LPCI) sides (shell side) of the 2B and 3A LPCI heat exchangers noted conductivity, chloride concentration, and sulfate concentration higher than normal torus water chemistry conditions caused by a tube leak in the LPCI heat exchangers.

Dresden and Quad Cities chemistry procedures require action to be taken to return an abnormal chemistry parameter to the desired range when that chemistry parameter is outside of its goal value (value or range, set below the action levels, that the plant is capable of achieving under normal conditions with good practices). The procedures require increased sampling to verify the effectiveness of corrective actions taken to address an abnormal chemistry condition.

Dresden and Quad Cities chemistry procedures require an evaluation be performed to determine if continued operation results in minimized degradation in the event a chemistry parameter reaches Action Level 2 (the value that represents the range outside of which data or engineering judgment indicate that significant degradation of the system may occur in the short term, thereby warranting a prompt correction of the abnormal condition). The chemistry excursion is then documented in a condition report in accordance with plant administrative procedures. The corrective actions program ensures that the conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined and a corrective action plan is developed to preclude repetition.

Evaluations that were conducted after the above chemistry excursions did not warrant any inspections and the problems identified were determined not to cause a significant impact to the material condition of the plants. Adequate corrective actions were taken to prevent recurrence.

Dresden and Quad Cities reactor water chemistry procedures require increased sampling to verify the effectiveness of corrective actions taken to address an abnormal chemistry condition.

<u>RAI B.1.3</u>

(a) The staff notes that NUREG-1801, in accordance with the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, specifies volumetric inspection for studs in place and both surface and volumetric examination of studs when removed. The applicant states in the LRA that, instead of a surface inspection, Dresden and Quad Cities use a VT-1 visual inspection, as granted under relief requests CR-13 and CR-11, respectively. Likewise, instead of a volumetric examination with a conventional UT, the Dresden and Quad Cities reactor closure head studs are examined by end-shot UT, as approved in relief request CR-12. Use of VT-1 visual inspection is acceptable based on current revisions of the ASME Code. However, use of the end shot UT inspection procedure was not approved per relief request CR-12 since it does not provide the required sensitivity (see Section 3.1.1.3 of the staff's SE dated 9/15/95). The staff's SE did approve the use of bore probe inspection procedure through the 3rd ISI interval. Future relief requests may be submitted by the applicant in accordance with 10 CFR 50.55a. Otherwise, the applicant must comply with the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, that specifies volumetric inspection for studs in place and both surface and volumetric examination of studs when removed. Please confirm that aging effects for the reactor closure head studs will be monitored/managed in accordance with the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1 for license renewal.

(b) The applicant states in LRA Appendix B.1.3, Reactor Head Closure Studs, that the reactor head studs at Dresden and Quad Cities are not metal plated and have had manganese phosphate coatings applied. Describe the D/QCNPS experience with the manganese phosphate coatings. Specifically, describe any cracking of the reactor head closure studs since the application of the manganese phosphate coatings.

(c) The applicant states in LRA Appendix B.1.3, "Reactor Head Closure Studs," that the reactor head closure studs management program provides for condition monitoring and preventive actions to manage stud cracking and loss of material. However, loss of material is not identified as an aging effect for reactor head closure studs in LRA Tables 3.1-1 or 3.1-2. Clarify this discrepancy and discuss D/QCNPS operating experience with respect to loss of material for the reactor head closure studs.

(d) The staff reviewed the UFSAR supplement to determine whether it provides an adequate description of the program. The UFSAR supplement should be revised to indicate that

VT-1 visual and bore probe UT inspection procedures are to be used for detecting loss of material and cracking in the reactor head closure studs.

Response:

(a) Since submittal of the LRA, Dresden and Quad Cities have updated their ISI Programs to be consistent with the 1995 Edition, 1996 Addenda of ASME Section XI. With this update, Dresden and Quad Cities removed the exception noted as Relief Request CR-12. However, it should be noted that the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, will be augmented by Code Case N-307-2 "Revised Ultrasonic Examination Volume for Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1, When the Examinations are Conducted From the End of the Bolt or Stud or From the Center-Drilled Hole", as endorsed by NRC Regulatory Guide 1.147, Revision 13.

(b) The studs were manganese phosphate coated as part of the original General Electric Purchase specification requirements. Four studs at Dresden Unit 2 were found to have cracking during refuel outages D2R11 (studs 47 and 60) and D2R15 (studs 52 and 81). These studs were replaced. No other recordable indications have been identified on the Dresden or Quad Cities reactor head closure studs.

(c) The loss of material was inadvertently added to LRA Appendix B.1.3 "Reactor Head Closure Studs" and should have been deleted. Dresden and Quad Cities do not have any operating experience that would indicate that loss of material is an applicable aging effect for reactor head closure studs.

(d) The UFSAR Supplement described in LRA Appendix A, A.1.3 provides a level of detail consistent with that provided in NUREG-1800, Table 3.1-2. Defining the specific examinations required by ASME Section XI is redundant to the stated information and is not required.

<u>RAI B.1.4</u>

The staff-approved version of BWRVIP-48 recommends enhanced VT-1 (EVT-1) for furnace-sensitized (from PWHT) welds, Alloy 182 welds, and the welds attaching certain components to the vessel. To facilitate staff's review, identify the D/QCNPS vessel ID attachment welds, weld materials, and the welds that are furnace sensitized. Also identify the attachment welds that will be inspected with enhanced VT-1.

Response:

BWRVIP-48 "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines" (Paragraph 3.2.1) recommends an enhanced VT-1 (EVT-1) for the core spray bracket welds and jet pump riser brace welds. Additionally, the BWRVIP recommends an EVT-1 for the steam dryer brackets and feedwater sparger brackets when the welds are furnace-sensitized stainless steel or are made with Alloy 182 filler materials. BWRVIP-48 (Table 3.2) recommends no additional inspections above those specified in ASME Section XI for the surveillance sample holder attachments. As indicated below, Dresden and Quad Cities examine the Vessel ID Attachment Welds as recommended by BWRVIP-48.

Attachment Weld	Weld Material	Furnace Sensitized	Inspection Method
Steam Dryer Support Bracket			
Dresden 2	E308	Yes	EVT-1
Dresden 3	E308	No	EVT-1
Quad Cities 1 & 2	E308	No	EVT-1
Steam Dryer Lower Guide Rod Bracket			
Dresden 2	E308	Yes	EVT-1
Dresden 3	E308	No	EVT-1
Quad Cities 1 & 2	E308	No	EVT-1
Core Spray Bracket			
Dresden 2	E308	Yes	EVT-1
Dresden 3	E308	No	EVT-1
Quad Cities 1 & 2	E308	No	EVT-1
Feedwater Sparger Bracket			
Dresden 2	E308	Yes	EVT-1
Dresden 3	E308	No	EVT-1
Quad Cities 1 & 2	E308	No	EVT-1
Jet Pump Riser Brace			
Dresden 2	E308, ER308, E308L, ER308L E308Si, E308LSi	No	EVT-1
Dresden 3	E308, ER308, E308L, ER308L, E308Si, E308LSi	No	EVT-1
Quad Cities 1 & 2	E308L, ER308L	No	EVT-1
Secondary Jet Pump Riser Brace, Double Leaf			
Exists at Dresden 3 Only	E308L, ER308L	No	EVT-1
Surveillance Sample Holder			
Dresden 2	E308	Yes	VT-1
Dresden 3	E308	No	VT-1
Quad Cities 1 & 2	E308	No	VT-1

RAI B.1.7

(a) The applicant credits BWR stress corrosion cracking AMP (LRA Appendix B.1.7) and water chemistry (LRA Appendix B.1.2) for managing cracking due to IGSCC in reactor vessel safe ends and reactor coolant pressure boundary piping. The BWR stress corrosion cracking AMP is based on BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection schedules." Please provide information regarding the plant-specific experiences related to IGSCC cracking of the reactor vessel safe ends and reactor coolant pressure boundary piping, mitigative actions taken, and the revised inspection schedules following the BWRVIP-75 guidelines to provide evidence that the AMP is effective. Also, provide information regarding whether hydrogen water chemistry and noble metal chemical application (NMCA) are implemented at D/QCNPS and how implementation has affected monitoring of water chemistry parameters. Response should include discussion relative to the reactor vessel safe ends as well as other components.

(b) UFSAR Supplement A.1.7 needs to be changed to make reference to NUREG-1801 XI.M7, "BWR Stress Corrosion Cracking."

Response:

(a) Exelon has reviewed the Dresden and Quad Cities operating experience related to IGSCC of the reactor vessel safe ends and reactor coolant pressure boundary piping. Reactor coolant pressure boundary piping was identified to have flaw indications on the reactor vessel safe ends and IGSCC on recirculation system piping. However, there were no flaw indications (IGSCC) identified that affected the component intended function for any components in the affected systems. The following are representative examples of IGSCC operating experience related to reactor vessel safe ends and reactor coolant pressure boundary piping. These examples demonstrate the effectiveness of the AMP.

- The IGSCC inspection of Quad Cities Unit 1 refueling outage Q1R15 in December 1998 identified some flaw indications on recirculation piping system that exceeded allowable values. Flaw evaluation and repair (weld overlay) were performed to justify continued plant operation. The recirculation piping is original piping and the associated IGSCC susceptible welds have received Induction Heat Stress Improvement (IHSI) treatment. The evaluation of the effectiveness of IHSI treatment for susceptible welds resulted in an adjustment of the inspection plan to change all Unit 1 28" IHSI treated Category C (non-resistant material, stress improvement after 2 years of unit operation) welds to Category D (non-resistant material, no stress improvement).

- IGSCC inspection during Dresden Unit 2 refueling outage D2R14 in June, 1995 found circumferential crack indications in two recirculation pipe welds. Flaw evaluations were performed to support continued plant operation without repair. The inspection plan was adjusted to re-inspect these welds every refueling outage.

- During Dresden Unit 2 refueling outage D2R02 in February 1972, RPV safe ends were inspected by VT, PT and UT. Several indications were identified and ground out until they were acceptable.

- During Dresden Unit 2 refueling outage D2R05 in September 1977, 27 sensitized safe ends were inspected. On Nozzle N9 (CRD return nozzle), the indications were determined to be unacceptable and the safe end was replaced with 316L. This replacement was one of a number of safe end replacements made for Unit 2 (reference UFSAR Table 5.2-4) prior to issuance of GL 88-01. The indication on Nozzle N2C (recirculation system inlet nozzle) was also unacceptable. It was ground out and a subsequent PT test was performed with acceptable results.

Mitigation actions taken to address intergranular stress corrosion cracking (IGSCC) cracking of reactor vessel safe ends and reactor coolant pressure boundary piping are as follows:

- Dresden and Quad Cities have replaced the RWCU system piping with piping that is resistant to IGSCC.

- Dresden Unit 3 has replaced the reactor recirculation system piping with piping that is resistant (316 NG with maximum contents of 0.02 wt % carbon and 0.10 wt % nitrogen) to IGSCC.

- Replacement stainless steel components are provided in the solution annealed condition, with carbon less than 0.035 wt % and ferrite levels greater than 7.5 wt %.

- Existing stainless steel weldments are treated with IHSI to minimize tensile stresses and provide mitigation of IGSCC. Alloy 82 is used for nickel base alloy filler material.

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- The Noble Metal Chemical (NMC) Injection system was installed to enhance the IGSCC mitigation. Decomposition of the compounds forms a thin layer (~1 $\mu g/cm^2$) of Pt and Rh, providing a catalytic surface on the reactor piping and internals. NMC injections have occurred at each site during outages in the 1999/2000 timeframe. No information is yet available on the effectiveness of the injections on IGSCC mitigation.

- The Hydrogen Water Chemistry (HWC) Injection system was installed to enhance the IGSCC mitigation by reducing the amount of oxidizing radiolysis products by injecting hydrogen into feedwater while maintaining the concentration of reactor coolant ionic impurities. HWC injection is conducted continuously (as much as practicable) during normal unit operation. Data from Dresden Unit 2 has indicated that IGSCC in the reactor recirculation piping can be suppressed by HWC injection along with control of impurity concentrations in reactor water.

- Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in EPRI TR-103515R2 to maintain high water purity.

Both Dresden and Quad Cities have implemented HWC and NMC injection. As part of the implementation of HWC and NMC, monitoring of electrochemical corrosion potential (ECP) was added. This is a measure of the voltage that arises when a metal is in contact with a solution. ECP is measured by comparison to a standard reference electrode at the temperature of interest. ECP data and HWC index results are used to calculate crack growth rate factors of improvement. BWRVIP modeling for BWR crack growth indicates that crack growth rate decreases with decreasing ECP.

At Quad Cities Station, Category C through E welds (Quad Cities currently has no Category B welds) are still being inspected to the same frequency as specified in BWRVIP-75, "BWR Vessel and Internal Project Technical Basis for Revisions of Generic Letter 88-01 Inspection Schedules" guidelines for normal water chemistry. Category A (resistant material) welds are fabricated with IGSCC resistant materials and are inspected per the Risk Informed In-Service Inspection (RISI) guidelines. Quad Cities has implemented the HWC and NMC addition. However, it has not reduced the inspection frequencies as specified in BWRVIP-75 and NRC SER of EPRI Report TR-113932 ("BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)"), dated May 14, 2002. Based on the effectiveness of the HWC and NMCA, the inspection frequencies may be adjusted in the future, which will follow the requirements of BWRVIP-75.

At Dresden Station, Category C through E (cracked, overlay, or stress improved) welds (Dresden currently has no Category B welds) are being inspected to the frequency specified in BWRVIP-75 guidelines either for normal water chemistry (NWC) or HWC/NMCA. Hydrogen water chemistry/noble metal chemical application inspection frequencies were reduced only for Unit 2 locations where the improved water chemistry is effective. Unit 3 maintains the normal water chemistry inspection frequencies. The hydrogen water chemistry system has been in use for Unit 2 since 1983. The system was not implemented for Unit 3 until 1996. Category A welds are fabricated with IGSCC resistant materials and are inspected per the RISI guidelines. There are no Category E welds on Dresden Unit 3.

The corresponding number of welds and frequency of inspection for Dresden Units 2 and 3 are provided in the following table. The schedule for Quad Cities has remained unchanged as indicated in the first sentence of the second paragraph above. The frequency changes for safe ends and other RPB piping components are included in the table below.

Category	Total Population	Welds Inspected
Unit-2:		
C-HWC	28	3 (10% every 10 years)
C-HWC	66	17 (25% every 10 years)
D-HWC	41	41 (100% every 10 years, 50% in first six years)
D-HWC	24	24 (10% every 6 years)
E-HWC	37	4 (10% every 10 years)
E-HWC	1	1 (25% every 10 years)
Unit-3:		
C-NWC	50	13 (25% every 10 years)
D-NWC	13	13 (100% every 6 years)

(b) Section 3.1.3.4 of NUREG 1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, requires the reviewer to verify that the applicant has provided information in the UFSAR supplement equivalent to that found in table 3.1-2. Information provided in UFSAR Supplement A.1.7 contains the exact wording found in table 3.1-2 of NUREG 1800 for the BWR Stress Corrosion Cracking program. Furthermore, there is no requirement to reference the BWR Stress Corrosion Cracking program described in NUREG 1801 XI.M7. As such, no changes are required to UFSAR Supplement A.1.7.

RAI B.1.8

According to BWRVIP-27, the ΔP/SLC nozzles at D/QCNPS are made of low-alloy steel instead of Alloy 600 and are susceptible to cracking. BWRVIP-27 describes an inspection strategy for these nozzles that includes volumetric inspection of the nozzle-to-shell weld and the nozzle inner blend radius at each inspection interval. In addition, NUREG-1801, Chapter XI.M8 requires the ΔP/SLC nozzles to be inspected in accordance with the requirements of ASME Section XI, Subsection IWB. In Appendix B.1.8 of the LRA, the applicant states that the Dresden and Quad Cities programs utilize relief request ISI CR-01 (relief granted per SER dated September 15, 1995) that provides for inspection of the inner blend radius by a VT-2 examination instead of the normal volumetric examination. Future relief requests may be submitted by the applicant in accordance with 10 CFR 50.55a. Otherwise, the applicant must comply with the appropriate requirements of the ASME Code. Please confirm that the aforementioned aging effects for the ΔP/SLC nozzles at D/QCNPS will be inspected in accordance with the requirements of ASME Section XI, Subsection IWB for license renewal.

Response:

Dresden and Quad Cities will inspect the ΔP/SLC nozzles in accordance with the requirements of ASME Section XI, Subsection IWB during the license renewal period, as part of the NRC approved ISI plan in accordance with 10 CFR 50.55a.

<u>RAI B.1.9</u>

(a) BWRVIP-26 "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines," states that the projected minimum end-of-life fluence at the grid beam location after 48 EFPY of operation is approximately 6×10^{20} n/cm^2 (E > 1 MeV), which is higher than the IASCC threshold of 5×10^{20} n/cm^2 (E > 1 MeV). Therefore, according to the staff final safety evaluation report for BWRVIP-26, one of the license renewal applicant action items is to identify and evaluate the projected accumulated neutron fluence as a potential TLAA issue. Confirm whether D/QCNPS follows the BWRVIP-26 guidelines for managing cracking in top guide due to IASCC. If so, then evaluate the projected accumulated neutron fluence as a potential TLAA issue. Also confirm whether the enhanced visual inspection technique EVT-1, recommended by BWRVIP-26, will be used to inspect the sites on the top guide that are likely to receive neutron fluence higher than the IASCC threshold before the end of the extended period of operation.

(b) One of the license renewal action items for BWRVIP-25 "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines," recommends that the applicant for license renewal should identify and evaluate the projected stress relaxation in the rim hold-down bolts as a potential TLAA issue. Confirm whether D/QCNPS follows the BWRVIP-25 guidelines for managing aging of the rim hold-down bolts. If so, then identify and evaluate in the LRA the projected stress relaxation in the rim hold-down bolts as a potential TLAA issue.

(c) The applicant states that the BWR vessel internals aging management activities have detected cracking in several vessel internals including Quad Cities access hole covers, and core spray piping at Dresden Unit 3. Discuss specific BWRVIP guidelines used to support the aging activities mentioned in LRA, Appendix B.1.9.

(d) In LRA Appendix B.1.9, the applicant reports that a jet pump beam assembly failed at Quad Cities Unit 1 in January 2002, and all similar beams have been replaced with the ones with improved heat treatment. Section 2.3.2.4 of BWRVIP-41 details mitigation processes that include a specific heat treatment that improves on the old heat treatment of the jet pump beams. Section 2.3.2.7 of BWRVIP-41 also recommends, for the improved heat treated beams along with reduced pre-load, inspections consisting of no inspection during the first 10 years of service and inspection of these beams every, following 10 year period with the same frequency as the old heat-treated beams with reduced pre-load. How do the new jet pump beams meet these BWRVIP-41 heat treatment guidelines and will they be inspected accordingly? Describe the beam assembly failure or provide a reference. Confirm whether all the beams at all four D/QCNPS units have been replaced with the ones with improved heat treatment.

Response:

(a) Dresden and Quad Cities are following the recommendations of BWRVIP-26 "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines," including the enhanced visual inspection technique EVT-1 of the top guide. IASCC of the reactor internals was evaluated as a potential TLAA and was determined to not be a TLAA.
However, Dresden and Quad Cities agree to perform inspections of the top guide similar to the inspections of the Control Rod Drive Housing (CRDH) guide tube. The inspection of the CRDH guide tube is performed in accordance with BWRVIP-47, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines. The examination extent and frequency is a 10% sample of the total population within 12 years, one-half (5%) to be completed within six years. The method of examination is EVT-1. LRA Appendix B.1.9, BWR Vessel Internals Program, will be enhanced to include inspection of top guide with examination extent and frequency similar to the CRDH guide tube. The program enhancements will be implemented prior to the end of the initial operating license term for Dresden and Quad Cities.

However, Exelon reserves the right to modify the above agreed upon inspection program should the BWRVIP-26 be revised in the future. This aging management inspection was accepted by the NRC staff in NUREG-1769 Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, in paragraph 4.5.2.

(b) Dresden and Quad Cities do follow the BWRVIP-25 "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines." . However, the Dresden and Quad Cities core plates had wedges installed with the shroud tie rod repairs. Therefore, inspection of the rim hold down bolts is not recommended by BWRVIP-25.

(c) The aging management activities associated with detecting these cracks associated with the core spray piping were based on BWRVIP-18 "BWR Core Spray Internals Inspection and Evaluation Guidelines". The examination methods included EVT-1. The access hole covers were inspected by VT-1 and VT-3 based on the recommendations of GE SIL 462 and Supplement 1 "Shroud Access Hole Cover Cracks". As a result of these inspections, cracks were found in the welded access hole covers for Dresden 2 and Quad Cities 1 & 2. Those access hole covers were subsequently replaced with mechanical covers (refer to the response to RAI 3.1-8). Future inspections of the Dresden 3 welded access hole covers will continue to be based on the SIL guidance.

(d) On January 9, 2002 jet pump beam 20 for Quad Cities Unit 1 failed due to an IGSCC crack in the transition area, which was a low stress area. For a more detailed explanation see Quad Cities Licensee Event Report LER 1-02-001. Subsequently, Dresden and Quad Cities replaced all of the jet pump beams with ones having improved heat treatment. The new beams are inspected based on the guidelines in BWRVIP-03 and BWRVIP-41.

RAI B.1.10

In LRA Appendix B.1.10, the applicant states that the component-specific evaluation for loss of fracture toughness in CASS vessel internals will be performed. The applicant further states that if loss of fracture toughness affects function of a given component, that component will be inspected as part of the D/QCNPS ISI program. Confirm that the criteria given in GALL AMP XI.M13 will be applied to determine whether loss of fracture toughness affects function of the CASS vessel internals. Also confirm that a supplemental inspection program that is qualified for detecting the critical flaw size with adequate margin will be provided for the CASS vessel internals whose function is affected.

Response:

Aging Management Program B.1.10 has been developed to evaluate thermal aging/neutron embrittlement of cast austenitic stainless steel reactor internals components that are included within the scope of license renewal. When implemented, the aging management program will be consistent with the program described in NUREG 1801 AMP XI.M13. For each component, the ferrite content will be determined based on Hull's equivalent factors (described in NUREG/CR-4513, Rev 1, Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems, U.S. Nuclear Regulatory Commission). Molybdenum content will be obtained from certified material test reports. Based on these factors, the potentially susceptible components will be identified. For these components, a mechanical loading assessment will be performed to determine maximum tensile loading on the component during ASME Code Level A, B, C and D conditions. For components that do not satisfy the acceptance criteria, an inspection will be performed as part of the ISI program. If any criteria are not met, a condition report will be generated for engineering evaluation.

The enhanced visual inspection program for detecting critical flaw size is in accordance with BWRVIP-03, which has the ability to achieve a 0.0005-in. resolution as specified in NUREG 1801 AMP XI.M13.

RAI B.1.11

It is noted that an aging management program for flow accelerated corrosion cracking has been used at Dresden and Quad Cities for many years and therefore experience should exist as to the effectiveness of the program to manage this type of aging degradation. Describe the experience at Dresden and Quad Cities with flow accelerated corrosion and the ability of the inspection programs to detect wall thinning in a timely manner before the intended function of piping components has been lost (have components been identified that did not meet the minimum wall predictions prior to replacement or loss of pressure retaining capacity?). What corrective actions have been taken, and to what extent have these measures been effective in eliminating or reducing the wall thinning? What changes to the program have occurred to ensure that flow accelerated corrosion has been successfully managed? Provide evidence that the current aging management program has been effective to successfully mitigate and detect wall thinning during the time period addressed by the LRA.

Response:

The flow accelerated corrosion (FAC) programs at both Dresden and Quad Cities have identified wall thinning prior to the loss of the intended function of the piping. The corrective actions have included:

- Replacing the localized (thinning) pipe sections with like-for-like piping.
- Replacing the localized (thinning) pipe sections with FAC resistant material.
- Replacing the entire piping run with either like-for-like or FAC resistant material.
- Evaluating the remaining wall thickness for continued use and re-inspection.

The following table provides a list of FAC-related Condition Reports (Operating Experience) and associated corrective actions over the last two years.

Unit	Date Identified	Condition Found	Corrective Action	CR / AR Number
D2	9/01/01	Main Steam leak found from valve body, weld or piping near PCV-2/3-3099-66A	Replace the line with FAC resistant material	74136
D2	11/06/01	FAC inspection found 2-3103-B2 Heater shell thinning	Install a plate patch repair	81864
D2	11/06/01	FAC inspection found wall thinning on line 2-3010-2"	Replace with FAC resistant material	81874
D2	11/07/01	FAC inspection found wall thinning on line 2-3407-2"	Replace with FAC resistant material	82062
D2	11/07/01	Potential FAC found as a result in change to MOV Program	The FAC issues addressed separately. Re-evaluate MOV program for proper stem lubrication	82043
D2	11/07/01	FAC inspection found wall thinning on line 2-3140-1½"	Replace with FAC resistant material	82055
Q2	2/14/02	FAC inspection found wall thinning on 2-3104-C1 feedwater heater	Perform weld build-up	95195
Q2	2/15/02	Wall thinning on 2C3 heater vent line found during pipe replacement	Sample similar areas and replace with FAC resistant material	95310
Q2	2/16/02	Wall thinning on 2-3105-D1 feedwater heater found	Replace with corrosion resistant material	95486
Q2	2/17/02	FAC inspection found wall thinning on line 2-3009A-1"	Replace with FAC resistant material	95554
Q2	2/18/02	FAC inspection found wall thinning on line 2-1SSH1-8"	Replaced degraded piping	95679
Q2	2/21/02	FAC inspection found wall thinning on low pressure turbine exhaust steam nozzle for line 2-3111-12"	Repair and install a FAC resistant liner	96093
Q2	2/24/02	Valve body for MO 2-3609-C found eroded	Replace with a manual valve	96518

Unit	Date Identified	Condition Found	Corrective Action	CR / AR Number
D3	3/15/02	Steam leak found on Extraction Steam nozzle of 3C2 feedwater heater	Repair using clamshell, replace shell and nozzle next refuel outage	99504
D3	10/15/02	3C2 feedwater heater shell eroded	Engineering evaluated acceptable for continued use, re-examine next refuel outage	127355
D3	10/16/02	FAC inspection found wall thinning on elbow of line 3-1SLMSV1-3"	Replace degraded material	127630
D3	10/16/02	FAC inspection found wall thinning on elbow of line 3-1SSH1-8"	Replace with FAC resistant material	127647
Q1	11/09/02	FAC inspection found wall thinning on bypass leak-off line	Replace degraded piping	130896
Q1	11/10/02	FAC inspection found wall thinning on line 1-3010-2" and 1-3011-2"	Replace with FAC resistant material	131009
Q1	11/11/02	FAC inspection found wall thinning on line 1-3623A-4"	Replace with FAC resistant material	131110
Q1	11/13/02	FAC inspection found wall thinning on extraction steam nozzle 1EA01D	Repair and installed a FAC resistant liner	131523
Q1	11/15/02	FAC inspection found steam seal headers degraded	Replace with FAC resistant material	131914
Q1	11/25/02	Turbine sealing steam drain has a leak	Replace with FAC resistant material	133153
Q1	4/23/03	HPCI drain line has a steam leak	Replace with FAC resistant material	155375

As seen in the above table, FAC-related degradation is usually identified prior to loss of the pressure retaining function of the component. The locations where leaks occurred prior to detection were in areas of low safety significance. Additionally, corrective actions include replacing the susceptible component with FAC resistant material when practicable, thereby increasing the effectiveness in eliminating or reducing FAC.

Program changes to improve effectiveness of the FAC Program have included:

- Maintaining or upgrading to the current state of the art versions of EPRI approved FAC modeling and predicting software (i.e. CHECKWORKS)
- Planned replacement of piping with FAC resistant material prior to identifying degradation
- When degradation is detected, such as predicted or actual less than the minimum acceptable wall thickness, additional evaluations or inspections are performed to bound the area of thinning

The FAC program is an inspection and monitoring program, not typically credited with mitigating functions. However, the planned replacement of susceptible piping with FAC resistant material is considered to be a mitigating action. The implementation of the FAC program using CHECKWORKS has proven to be effective in prediction and detection flow accelerated corrosion.

RAI B.1.12

(a) LRA Table 3.4-1, item 3.4.1.3 states that the external surfaces of carbon steel components in the SPC systems are managed for loss of material due to general corrosion as described in LRA Section 3.4.1.1.3. LRA Section 3.4.1.1.3 states that, "aging management of the external surface of the main steam, feedwater, condensate and condensate storage system components in a sheltered environment with moist, warm air will be managed either by the Structures Monitoring Program (B.1.30) or by system engineer walkdowns performed by the Bolting Integrity Program (B.1.12) aging management activities."

As described in Section B.1.12 of the LRA, the applicant's Bolting Integrity Program consists of visual inspections for loss of material for bolting, but does not address system walkdowns to inspect external surfaces of all carbon steel components in the SPC systems.

Since the GALL report recommends aging management for loss of material due to general corrosion for external surfaces of all carbon steel structures and components, explain how the Bolting Integrity Program provides aging management for loss of material due to general corrosion for external surfaces of all carbon steel SPC system piping and components in an environment of air, moisture, and temperature less than 212 EF, or provide an alternate plant specific program to manage these aging effects.

(b) The LRA takes exception to the NUREG-1801 AMP XI.M18 program scope element which states that the Bolting Integrity Program covers all bolting within the scope of license renewal including structural bolting. The LRA states that the aging management of structural bolting at Dresden and Quad Cities will be addressed in the Structures Monitoring Program (AMP B.1.30). However, AMP B.1.30 does not include any discussion describing how the applicant intends to manage structural bolting integrity relative to the recommendations delineated in NUREG-1801 AMP XI.M18. Provide additional information regarding AMP B.1.30 to include a discussion describing how the aging management of structural bolting integrity will be performed relative to the recommendations in NUREG-1801 XI.M18.

(c) The LRA takes exception to NUREG-1801 reference to industry consensus recommendations in EPRI TR-104213 as a bolting integrity aging management program basis for nonsafety related bolting. The applicant goes on to state that the Dresden and Quad Cities programs address the guidance contained in EPRI TR-104213 but do not specifically cite its use. It is not clear to the Staff exactly what the applicant is requesting with this exception. The Staff would expect that, consistent with the recommendations in NUREG-1801 AMP XI.M18, the applicant's bolting integrity aging management program for nonsafety related bolting would meet or exceed the standards delineated in EPRI TR-104213. Please provide clarification.

(d) For safety related bolting, the bolting integrity program described in NUREG-1801 XI.M18 relies on the NRC recommendations and guidelines for comprehensive bolting integrity programs that are delineated in NUREG-1339 and the industry's technical basis for the program and guidelines for material selection and testing, bolting preload control, inservice inspection (ISI), plant operation and maintenance, and evaluation of the structural integrity of bolted joints, outlined in EPRI NP-5769, with the exceptions noted in NUREG 1339. The LRA states that the bolting integrity program at Dresden and Quad Cities incorporates industry recommendations addressed in EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," yet makes no reference to the NRC recommendations delineated in NUREG-1339 and NRC exceptions to EPRI NP-5769. Provide additional information regarding the applicant's position relative to the NRC recommendations delineated in NUREG-1339 including NRC exceptions to EPRI NP-5769.

(e) The LRA takes exception to the NUREG-1801 XI.M18 provision to include, where applicable, periodic monitoring for loss of preload and states that the Dresden and Quad Cities programs do not include inspections for loss of preload because loss of preload in a mechanical joint is a design driven effect and not an aging effect.

Whether or not the joint is going to perform its intended function depends to a large extent upon the preload in the bolts and the resulting clamping force on the joint interface. Although many of the design factors contributing to joint design preload requirements are well known and can be adequately addressed (with consideration for uncertainties) in the design process, there are time-related operational conditions which can result in a significant reduction in preload that may not be well quantified. Many of the joints that are subjected to cyclic loads, especially large loads, will embed, and, therefore will relax more than joints under static loads. Vibration, flexing of the joint, cyclic shear loads, thermal cycles, and other factors can cause whole or partial self-loosening of a fastener. Depending on the application, preload reduction can contribute directly to material loss, fatigue and stress corrosion cracking. Consequently, the staff believes that a comprehensive bolting integrity program should, where applicable, include some periodic preload monitoring/checks of selected components/structures. Therefore, provide sufficient justification for the exception from the periodic loss of preload monitoring recommendation in NUREG-1801 XI.M18.

(f) The LRA states that the Dresden and Quad Cities bolting integrity programs do not address Class 1 NSSS component support bolts. Aging management of ASME Section XI Class 1, 2, and 3, and Class MC support members, including mechanical connections, is covered by the ASME Section XI, Subsection IWF (B.1.27) aging management program. The ASME Section XI bolting inspection requirements are specified in Table IWF-2500-1 and require VT-3 examinations. NUREG-1801 XI.M18 states that the bolting integrity aging management program monitors the effects of aging on the intended function of closure bolting, including loss of material, cracking, and loss of preload. VT-3 inspections evaluate the general mechanical condition of the bolting and can identify the presence of corrosion/material loss, but the VT-3 inspections cannot be relied upon to identify the presence of cracking or preload loss. Explain how these aging effects will be managed.

Response:

a) The Bolting Integrity program (B1.12) consists of visual inspections, which rely on detection of visible leakage during preventive maintenance and routine walk downs (routine observation activities). The walk downs are also credited for detecting aging degradation on the external surfaces of system piping and components. The Bolting Integrity program credits the routine walk downs as routine observation activities, which detect aging degradation on the external surfaces of system piping and components.

The routine walk downs include steam and power conversion systems in an environment of air, moisture, and humidity less than 212°F to manage loss of material due to general corrosion for external surfaces of all carbon steel components.

In addition, the Structural Monitoring program (B.1.30) is credited to manage aging of external surfaces of carbon steel components in the Steam and Power Conversion systems to manage loss of material due to general corrosion. The Structural Monitoring program consists of visual inspection of piping and components by area rather than by systems.

b) The aging management of structural bolting integrity is performed in the Structural Monitoring Program (B.1.30). Accessible bolted connections are visually inspected for loose or missing bolts or nuts or loss of material due to environmental corrosion. Anchor bolts are visually inspected for corrosion, missing and loose parts. Positive hold-down of the anchor bolt is verified by ensuring all faying surfaces are in contact.

c) Non-safety related bolting addressed in LRA Section B.1.12 meets the intent of the aging management attributes delineated in EPRI TR-104213. One of the exceptions to the bolting integrity program (B.1.12) is that the bolting program implementing procedures do not specifically reference EPRI TR-104213. However, the attributes for aging management of non-safety related bolting includes material procurement, use of approved lubricants and sealants, proper torquing, and leakage evaluations as specified in EPRI TR-104213. Exelon will enhance the implementing procedures for this aging management program to reference EPRI TR-104213. Maintenance evaluations and repairs of non-safety related bolted connections follow the EPRI bolting guidelines per EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and TR-113859 "Proceeding of the 1st International Conference on Sealing Technology and Plant Leakage Reduction (ICSTPLR-99)" for the evaluation and repairs of the flange and bolts. Component external surface degradation (loss of material) is inspected in the routine walk downs.

d) The bolting integrity program (B.1.12) at Dresden and Quad Cities meets the intent of EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and the exceptions noted in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," for performing material selection and testing, In-Service Inspection (ISI), and plant surveillance and maintenance practices. The corporate and station implementing procedures specifically cite EPRI NP-5769, however NUREG-1339 is not specifically referenced. Exelon will enhance the implementing procedures for this aging management program to reference NUREG-1339.

e) Exelon will manage the loss of preload for closure bolting in the reactor vessel system, recirculation pumps, reactor recirculation valves, reactor vessel head vent valves, and the reactor pressure boundary portion of all other systems. Aging management program, B.1.12, Bolting Integrity will be enhanced to include periodic inspections the closure bolting in accordance with the ASME Code Section XI requirements. Closure bolting will be periodically inspected for signs of leakage. The enhanced Bolting Integrity aging management program will be comprised of periodic In-Service Inspection (ISI), piping and component Preventive Maintenance inspections, and routine walk downs. These activities will detect early leakage and material degradation of closure bolting (that may be caused by loss of material or cracking) prior to loss of system or component intended functions. Periodic In-Service Inspection of closure bolting was accepted by the NRC staff as an acceptable aging management program for loss of pre-load for the components discussed above in NUREG-1769, Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, Section 3.1.3.2.1.

f) LRA aging management program B.1.27, ASME Section XI, Subsection IWF, manages the aging effects of ASME Section XI Class 1, 2, and 3, and Class MC supports members, including mechanical connections. LRA program B.1.27 meets the requirements delineated in NUREG 1801, XI.M18 for Class 1 NSSS components support bolts. As stated in NUREG 1801, XI.M18, structural bolting both inside and outside containment will be inspected by visual inspection. If the bolting is found corroded, a closer inspection will be performed to assess the extent of corrosion.

RAI B.1.22

The applicant plans to enhance the current reactor vessel surveillance program by making it consistent with the staff-approved versions of BWRVIP-78 and BWRVIP-86. The staff has concluded that the final proposed BWRVIP integrated surveillance program (ISP) was acceptable for BWR licensee implementation for the current term as documented in the SER attached to the letter from Bill Bateman, NRC, to Carl Terry, BWRVIP Chairman, dated February 1, 2002. One of the provisions of the ISP is for surveillance capsule material withdrawal and testing during the license renewal period. A revision to BWRVIP-78 and –86 reports to include license renewal is in progress and will be submitted to the NRC for review in the near future. The applicant must commit to incorporate the reactor vessel surveillance program consistent with the staff-approved versions of the revised BWRVIP-78 and BWRVIP-86 documents and include this commitment in the UFSAR supplement for this program.

Response:

Section B.1.22 of the LRA states that the Exelon reactor vessel surveillance program will be enhanced to incorporate the reactor vessel surveillance vessel surveillance program consistent with the staff-approved versions of BWRVIP-78 and BWRVIP-86. This commitment is already included in section A.1.22 of the Dresden and Quad Cities UFSAR supplement.

RAI B.1.24

(a) The Dresden and Quad Cities programs for selective leaching determinations include only one element of the method indicated in NUREG-1801, which specifies a one-time visual inspection and hardness measurement of a selected set of components. The Dresden and Quad Cities programs provide for visual examination and reject hardness testing. The applicant's justification is that baseline hardness values are not available for materials in the plants. Since materials typically have a normal or expected hardness range or baseline hardness testing could be performed in areas not susceptible to selective leaching, justify why deviation from a normal/expected hardness range could not be considered a useful indicator of selective leaching. If the measurement deviations were marginal, further characterization could be justified, (i.e., calibration using unleached and leached materials. For the visual examination method, what basis and criteria will be used to train personnel for identification of selective leaching and how will this be incorporated into the program? Is the uncertainty of visual inspection more or less than that inherent in the hardness method?

Selective leaching is indicated to often occur under deposits and in other nonvisible locations (Jones, Principles and Prevention of Corrosion, Macmillan, New York, 1992, pp. 19,20). What are the criteria for selecting sampling locations to assure a representative sample of components?

GALL Section XI.M33 expresses reservations regarding the effectiveness of the visual method. Given the reservations indicated, justify why the visual method is regarded to provide a reliable basis to evaluate selective leaching in Dresden and Quad Cities plants.

(b) What has been the operating experience with occurrences of selective leaching at Dresden and Quad Cities? What if any corrective actions have been taken in response to aging effects from this degradation mechanism? How will such experience be factored into applicants programs to detect and manage selective leaching?

(c) UFSAR Supplement A.1.24 needs to be changed to make reference to NUREG-1801 XI.M33, "Selective Leaching of Materials" and the use of visual supplemented by other examinations in lieu of hardness tests.

Response:

a) There are several reasons why Exelon does not believe that hardness testing of components for selective leaching is the preferred over visual testing. These are listed below:

- Different baseline hardness numbers between components
- Difference in physical geometry (thicker v/s thinner component etc.)
- Hardness testing tools may be different due to different geometry
- Normal variation in product hardness
- Surface condition variation
- Non-homogeneity of material
- Hardness allowable variation
- Cooling rate will affect the hardness due to geometry difference

There are many different grades of aluminum, cast iron, brass and bronze. In many cases, the original purchase order associated with the installed component did not specify the grade. The differences in grades also provide a wide variance in hardness.

In most instances, the internal component is not large enough to accommodate a hardness tester without destroying the component. Since the hardness method also requires engineering judgment considering the above listed variances, visual inspection performed by a VT qualified inspector is more certain than the hardness method.

The visual inspection will be performed consistent with ASME Section XI VT-1 visual inspection requirements. General visual inspection technique will be utilized to determine the condition of the component or surface. The qualification requirement will be added to the station work order to perform the work. Since baseline information is not available, visual inspection of a susceptible component by a VT qualified inspector provides reasonable assurance and identifies the evidence of selective leaching components.

The inspector will inspect for degradation of the component surface including plug type or localized and uniform dezincification indication. Dezincification (loss of zinc from brass and bronze) and graphitization (removal of iron from cast iron) are examples of such a process. A weak, porous, or spongy layer of the de-alloyed material frequently provides evidence of dezincification. In copper alloys, this will frequently have reddish color.

The scope of the selective leaching susceptibility applies to gray cast iron, brass, bronze, and aluminum bronze components in wetted environments. A sample of ten (10) components will be selected from wetted environment representing at least one component of each material type susceptible to selective leaching at each station. When visual inspection reveals evidence of selective leaching, the inspector may then use other approved inspection or testing methods (e.g. Ultrasonic Testing or mechanical measurements) to determine the remaining wall thickness of the sound metal. As another option, the component may be removed from the system and sent for microscopic examination to determine the amount of degradation. Any component analyzed to have an end of life less than the time remaining in the extended period of operation will be considered a "Failure" and will be replaced before actual failure occurs. Failure will be documented on a Condition Report in accordance with the Corrective Action Program. When a failure is identified, the sample population will be expanded to include additional components in the same environment, material, and preferably service condition as the noted failure. Samples will continue to be expanded until an acceptable population is obtained or all components in the failed material/environment for that station have been examined.

As stated above, visual inspection method will provide reasonable assurance and a reliable basis to evaluate selective leaching at Dresden and Quad Cities.

b) The selective leaching of materials aging management program is new. Any degradation of components due to selective leaching may have been classified with different aging mechanism and the component deficiency should have been corrected. No programmatic operating experience is available at Dresden and Quad Cities.

c) The following statement should have been included in the Dresden and Quad Cities UFSAR Supplement A.1.24.

The selective leaching of materials aging management program includes numerous one-time inspections of components of the different susceptible materials selected from each of the applicable environments to determine if loss of material due to selective leaching is occurring. These inspections will consist of visual inspection consistent with ASME Section XI VT-1 visual inspection requirements. If selective leaching is occurring the program requires evaluation of the effect it will have on the ability of the affected components to perform their intended functions for the period of extended operation, and of the need to expand the test sample. For systems subjected to environments where water is not treated (i.e., the open-cycle cooling water system) the program also follows the guidance of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

RAI B.1.25

(a) For the one-time internal UT inspections of buried steel tanks at Dresden and Quad Cites, will all buried tanks at both plants be inspected. If not, how will tanks considered to represent worst-case age-related degradation be selected?

Will 100% of surfaces of selected tanks be inspected? If not, what surfaces will be inspected, what will be the basis for selecting the internal surfaces to be inspected and how will the selected areas/sizes ensure that material degradation will be identified (i.e., will a grid be used to define the surface locations, what grid size and basis for the size and locations would be used to detect areas of localized degradation)? What basis will be used to ensure the chosen areas are representative of the worse case degradation.

What are the acceptance criteria for the UT inspections and what actions will be taken if the tank wall thicknesses are outside the acceptance criteria?

The ID surface examinations will detect degradation on interior surfaces. How will degradation (e.g., general, pitting, crevice, MIC) on exterior surfaces be assessed?

(b) As part of the buried piping and tank AMP, the program includes a one-time UT of the bottom of an outdoor aluminum storage tank at either Dresden or Quad Cities. It is not clear to the staff how the proposed buried piping and tank AMP and degradation mechanisms associated with this AMP are relevant to an above ground aluminum tank. It would appear to the staff that an enhancement/exception to the Carbon Steel Above Ground Tank AMP would be more relevant. Explain the correlation between degradation of the buried pipe and tank materials with degradation of the aluminum tank bottom. In addition, since (as stated in Table 3.4-2 of the LRA) NUREG-1801 does not address outdoor aluminum storage tanks resting on the ground, explain why including proposed one-time inspection for the above ground aluminum tanks as an enhancement to the buried piping and tank AMP is appropriate. Also, the applicant needs to explain how the tank will be selected and why the inspection of one tank at either Dresden or Quad Cities will be representative of the soil-to-tank bottom interactions for all aluminum tanks at both plant sites. Finally, please describe what are the acceptance criteria for the UT inspection and what actions will be taken if the tank wall thickness is outside the acceptance criteria (e.g., will additional tanks be inspected?).

(c) NUREG 1800 Section A1.2.3.10 indicates that the information provided by the operating experience of an AMP may indicate when an existing program has succeeded and when it has failed in intercepting aging degradation in a timely manner. Accordingly, an existing AMP is considered effective if operating experience, including the corrective actions, demonstrates that aging degradation can be found in a timely manner prior to the actual loss of the component intended function.

Operating Experience acknowledges that failures have occurred in Dresden/Quad Cities underground pipes. In light of these failures no explanation is provided as to why the existing AMP can be considered effective. Please discuss the root causes for these failures and indicate how the known causes are guiding the Buried Piping and Tanks Inspection program to assure that integrity of the entire underground inventory is being adequately addressed. For example, microbiologically influenced corrosion (MIC) is a highly stochastic degradation mode that may need special measures if there is evidence that it has been a factor in any of the pipe failures. If galvanic factors were involved, describe any changes or enhancements factored into the inspection program? Describe how operating experience has been factored into the inspection program.

(d) NUREG 1800 states that one-time inspections are employed to provide additional assurance that either aging is not occurring or the evidence of aging is so insignificant that an aging management program is not warranted for the period of extended operation. The LRA states that a one-time visual inspection involving the external surface of a section of buried fire main piping will be performed. Operating experience at Dresden and Quad Cities identified several failures in the fire main piping requiring the evacuation and repair of the piping. This would suggest that reliance on a one-time inspection would not be appropriate. Please explain why a one-time inspection is appropriate for the fire main buried piping. Also, identify how the piping sections will be selected and why they are representative of the most likely locations for pipe degradation.

(e) The GALL report indicates that underground pipes and tanks inspected or removed for any reason should be assessed as an element of the Buried and Tanks Inspection AMP. While the applicant alludes to inspection of underground components uncovered during maintenance, confirm that systematic assessment of underground pipes and tanks for age-related degradation during inspection or after removal will be an active element of the AMP. The GALL guidance calls for inspections in areas with the highest likelihood of corrosion problems. Confirm that Operating Experience and assessment of corrosion-prone locations (e.g., areas of water accumulation) will prompt inspections for timely detection of age-related degradation before loss of function occurs.

(f) Coatings and wrappings are indicated by the applicant as enhancements to the AMP. The GALL report identifies coatings and wrappings as an element of the AMP. Explain why coatings and wrappings are regarded as an enhancement in the proposed AMP.

(g) Pressure and leak tests and above ground walkdowns are proposed by the applicant as aging management methods. Explain how these are effective aging management practices when they seem instead to only detect aging degradation in advanced stages.

Response:

(a) Only one buried steel tank per site will be inspected. The affected tanks are the emergency diesel generator and SBO diesel generator fuel oil storage tanks. Worst-case tank selection will be based on input from site personnel, maintenance and chemistry sampling history of the tank, age of the tank, and its accessibility.

Only the bottom half of each of the selected tanks will be inspected. The bottom sections were chosen because fuel contaminants and degradation products are prone to settling to the bottom of the associated tanks. The basis for selection of the specific internal surfaces will be as determined by engineering personnel and will include locations where corrosion may be expected to occur, such as welds and low spots. A grid will be used to define the surface locations. The grid size will be as defined by appropriate station and corporate UT inspection procedures and as determined by engineering personnel. The basis for ensuring that the chosen areas are representative of the worst-case degradation will be as determined and documented by engineering personnel.

The acceptance criteria for the UT inspections will be as defined in appropriate station and corporate procedures, and as determined by engineering personnel, the basis of which will be ASME codes and ASTM standards. Actions to be taken in the event wall thicknesses are outside acceptance criteria will be:

1) Expand sample area and/or sample population.

2) Evaluate inspection results.

3) Implement necessary repairs.

EPRI 1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3, Appendix E, considers the soil and groundwater environment to be an aggressive corrosive environment for carbon steel components. It indicates that the applicable aging mechanisms are general corrosion, pitting corrosion, crevice corrosion and MIC. Degradation of exterior surfaces of the tanks due to these aging mechanisms will be assessed by:

1) Inspection of tank exterior coatings whenever the tanks are uncovered during station excavation activities.

2) Periodic tank leak testing and internal inspections to detect thru-wall degradation (pinholes) from the outside.

3) UT inspection of the bottom portion of the tank to detect wall thinning.

(b) The proposed buried piping and tank AMP and associated aging mechanisms are not relevant to an above ground aluminum tank. This AMP was inadvertently identified instead of the correct AMP, which is the Aboveground Carbon Steel Tanks AMP.

The LRA should have included the following changes:

1) LRA Section B.1.20, Aboveground Carbon Steel Tanks, should have referenced the UT inspection requirement for the associated above ground aluminum tanks. Since the AMP for aboveground carbon steel tanks does not include aluminum as a material type, an exception statement to this effect should have been included in this section.

2) LRA Section B.1.25, Buried Piping and Tanks Inspection, should have removed reference to the UT inspection requirement for the associated above ground aluminum tanks.

3) Item 3.4.2.42 of LRA Table 3.4-2 should have referenced the aboveground carbon steel tanks AMP.

4) LRA Section A.1.20 (for Dresden and Quad Cities) should have referenced the UT inspection requirement for the associated above ground aluminum tanks.

5) LRA Section A.1.25 (for Dresden and Quad Cities) should have removed the reference to the UT inspection requirement for the associated above ground aluminum tanks.

Tank selection will be based on input from site personnel, maintenance history of the tank, age of the tank, and its accessibility. The selected tank will be one of the affected Quad Cities Station tanks because the bottoms of the corresponding tanks at Dresden

Station have been recently (1992/1993 timeframe) replaced. Replacement of the tank bottoms at Dresden was necessary due to corrosion.

The results of the inspection on the selected Quad Cities tanks will be representative of those at Dresden Station because:

1) All the tanks are made of aluminum.

2) All the tanks are of similar construction.

3) The Quad Cities Station tank bottoms are older than those at Dresden Station and, therefore, more likely to show the effects of aging and gage the rate at which any noted degradation is occurring.

4) All the tanks have similar internal and external environments.

The acceptance criteria for the UT inspection will be as specified in station and corporate procedures and as determined by engineering personnel.

Actions that Exelon will take will be based on the actual conditions at the time and the requirements of the corrective action program.

(c) The failures identified in LRA Section B.1.25 were associated with the fire protection system and the demineralized water system at Dresden. The failures were attributed to the aging effect of loss of material. However, no specific aging mechanisms were identified. For those in-scope buried components having locations where dissimilar materials are used, the materials are steel and cast iron which have the same galvanic potential. Furthermore, the failed piping in the fire protection system was made of a non-metallic material (concrete asbestos), and the piping has been replaced with PVC. Therefore, it is apparent that these failures were not due to galvanic corrosion.

The fire water system flow test procedure for each site is being enhanced to include heightened awareness in the acceptance criteria that minor pressure losses during performance of the periodic flow tests may be indicative of minor leakage (pinholes) even though acceptable flow and pressure are present. Flow testing is performed every 3 years. This frequency is sufficient to allow detection of pipe degradation prior to loss of system function.

With respect to systems other than the fire water system, no other operating experience has been noted as having been specifically factored into the buried piping and tanks program. However, loss of function of those systems is unlikely, because degradation of each system would first be detected by periodic system monitoring (of flow, pressure indication) or functional testing.

(d) NUREG 1801 recommends that the buried piping and tanks program include periodic inspection to manage the effects of corrosion on the pressure retaining capacity of buried carbon steel piping and tanks. However, station history for Dresden and Quad Cities indicates that buried piping and tanks have been infrequently uncovered during station yard excavation work. Therefore, other inspection and testing activities were recommended to ensure that adequate aging management was provided for buried piping and tanks within the scope of LR. One-time inspection of a section of buried fire main piping was among the alternative activities recommended. Despite the fact that operating experience at Dresden and Quad Cities has identified several failures in the fire main piping, a one-time inspection is appropriate for this piping because:

1) The inspection will include provisions for expanding the sample size in the event acceptance criteria are not met.

2) The inspection will include provisions to establish root causes and corrective actions, including the possible creation of periodic inspections of buried piping at susceptible locations in the event acceptance criteria are not met.

3) The inspection will be performed at a location(s) determined to be representative of the most likely locations for pipe degradation.

4) The referenced failed piping in the fire protection system was made of a non-metallic material (concrete asbestos). That piping has been replaced with PVC.

In addition, the inspection of the external surfaces of buried in-scope components whenever they are uncovered during station excavation activities, though infrequent, remains a requirement of the program.

Piping section selection will be based on input from site personnel, piping maintenance histories, piping age, and its accessibility.

(e) Systematic assessment of underground pipes and tanks for age-related degradation during inspection or after removal will be an active element of the Buried Piping and Tanks Inspection AMP. The requirement for this assessment is contained in the corporate procedure governing structures monitoring. Specifically, this procedure requires inspection of exposed components for leakage, corrosion, coating degradation, misalignment of joints, and degraded/missing nuts or bolts at flange locations. In addition, site system engineers maintain system notebooks detailing failures on their assigned systems. Information contained in these notebooks, in combination with requirements identified in the structures monitoring procedure, will ensure that corrosion prone locations are inspected and that timely detection of age-related degradation occurs before loss of function.

(f) Necessary changes to the program are specifically identified in the "Enhancements" portion of the LRA Section B.1.25. Coatings and wrappings are not identified in this portion of LRA Section B.1.25. Coatings and wrappings are part of the existing program. The "Description" and "Exceptions to NUREG-1801" portions of LRA Section B.1.25 indicate that the buried piping and tanks inspection program "as enhanced" includes piping and component coatings and wrappings among other activities. These statements are not meant to imply that coatings and wrappings are an enhancement.

(g) As indicated in (d) above, inspection and testing activities were recommended as alternatives to the NUREG 1801-recommended periodic inspections of the external surfaces of in scope buried components. Pressure and leak tests and above ground walkdowns were among the alternative activities recommended. These activities are part of a comprehensive approach to managing aging of the in-scope buried components. They are not meant individually to provide total management of the aging of the components. Leak testing is to be performed periodically and will be an effective means to detect minor leakage from in scope piping and tanks in time for necessary repairs/replacements to be completed and prior to loss of function. Walkdowns of ground areas above in-scope buried components are also to be performed periodically and, to a lesser degree, are effective at detecting component degradation prior to loss of function. These walkdowns are used to detect degradation as evidenced by seepage or settling of the soil above the subject components. Although loss of function is a possibility prior to detection of a failure by walkdown, it is unlikely because system degradation would first be detected by periodic system monitoring (of flow, pressure indication) or functional testing.

RAI B.2.6

(a) Table 3.2-2 and Table 3.3-2 identify several local corrosion mechanisms for the tube side of the Quad Cities battery/station blackout room heat exchanger and Dresden and Quad Cities HPCI lubricating oil coolers and gland seal condensers. These include galvanic corrosion, crevice corrosion, pitting, MIC, FAC, selective leaching, etc. B.2.6 does not adequately define the specific examination/inspection activities that will be relied upon to detect these corrosion mechanisms nor does it justify why the proposed program activities can be expected to reliably detect the presence of these aging mechanisms before they impact the ability of the heat exchanger to perform its intended function(s). Describe the examination activities and the expected ability of these activities to detect aging effects of concern.

(b) B.2.6 indicates that past operating experience has shown that loss of material, cracking, and buildup of deposits in heat exchangers have been detected in Dresden and Quad Cities heat exchangers prior to loss of system intended functions. Please provide additional details regarding these occurrences including the heat exchanger, type of degradation mechanism, how it was detected, and corrective action taken, etc.

(c) The LRA states that specific acceptance criteria are provided in the inspection or test procedures but does not identify what evaluation methods and corresponding acceptance criteria/standards will be used. The applicant also states that EPRI guidance will be used to determine allowable percent wall loss, plugging criteria, and projections for remaining life but fails to identify the EPRI document(s) that will be used and their applicability to the degradations mechanisms and components addressed by this AMP. Provide a full reference to the EPRI report referred to in the LRA. Provide additional details regarding the acceptance criteria being employed against which the need for corrective action will be evaluated for the aging mechanisms and components in this AMP. Also, describe the methodologies that will be used to analyze the examination results against the applicable acceptance criteria.

(d) The LRA does not discuss how the program will monitor and trend cracking and material loss inspection results. Please provide additional details describing the methods that will be used to evaluate inspection results and assess remaining component life predications for applicable material loss and cracking mechanisms.

(e) The applicant identifies build up of deposit due to fouling as an applicable aging effect for the stainless steel tubes in the isolation condenser heat exchangers (Dresden only) exposed to steam on the tube side and demineralized water on the shell side. The applicant points out in the LRA Table 3.1-2, Ref. No. 3.1.2.15, that NUREG-1801 does not identify fouling as an applicable aging effect and refers to an "EPRI/SANDIA" report that identifies fouling as an applicable effect due to construction and operating conditions.

Provide a full reference to the EPRI/SANDIA report referred to in the LRA.

Summarize the industry and plant-specific experience related to fouling of the isolation condenser heat exchangers in demineralized water.

Response:

(a) Galvanic corrosion, crevice corrosion, pitting, MIC, and FAC for the HPCI lubricating oil coolers and gland seal condenser tube internal surfaces will be identified by periodic visual inspections and eddy current testing. The procedure that governs the visual inspections includes inspections of accessible tube internal surfaces. Eddy current testing is recognized by the NUREG-1801 as an effective method of measuring surface condition and the extent of wall thinning for heat exchanger tubes.

Selective leaching for the HPCI lubricating oil coolers and gland seal condenser tube internal surfaces will be identified by one-time visual inspections and, if necessary, microscopic examination of affected components. The one-time inspection will be performed on a component with a similar material and environment combination, but will not necessarily include the tubing of these heat exchangers. Visual inspection for selective leaching will be performed consistent with ASME Section XI VT-1 visual inspection requirements. This aging effect does not generally cause changes in dimensions and is, therefore, difficult to detect. However, visual inspection can be used to provide preliminary indication. Associated indication is frequently evidenced by a weak porous or spongy layer of the de-alloyed material. Engineering evaluations will be performed for any such indications accompanied by microscopic examination of the suspect components. Therefore, the program provides an effective means of detecting loss of material for the subject heat exchanger tubes.

Loss of material for the Quad Cities battery/station blackout room heat exchanger (air handling unit and condensing unit) tube internal surfaces will be identified by visual inspections of the accessible surfaces. No eddy current inspections are provided for this component. However, the procedure that governs this inspection contains provisions to initiate a work request to perform cleaning of the tubes in the event tubes need cleaning.

The LRA does not include selective leaching as an aging mechanism for the Quad Cities battery/station blackout room heat exchanger.

(b) The Heat Exchanger Test and Inspection Activities Program is a new aging management program. No operating experience exists at this time. However, similar controls have been implemented for the GL 89-13 Program heat exchangers. Operating experience associated with below-listed heat exchangers was identified by the GL 89-13 Program:

- RBCCW heat exchangers
- TBCCW heat exchangers
- ECCS Room cooler heat exchangers
- LPCI heat exchangers (Dresden only)
- Diesel generator cooling water pump heat exchangers
- RHR seal and motor coolers (Quad Cities only)
- Isolation condensers (Dresden only)

Examples of the types of degradation, methods of detection, and associated corrective actions identified for the above-listed heat exchangers includes:

- Cracking of LPCI heat exchanger heads detected by PT or UT as part of periodic inspections. Corrective actions included evaluation of inspection results, laboratory examination of material samples, grinding out indications, weld repair, and re-inspection.

- Loss of material for TBCCW tubing identified by eddy current testing as part of periodic inspections. Corrective actions included evaluating inspection results and replacing tubes with minimum wall thicknesses not meeting acceptance criteria.

- Buildup of deposits for ECCS room cooler components identified by cooler flow surveillances or operator rounds instrumentation inspections. Corrective actions included cleaning and subsequent inspection of surfaces.

- Loss of material of ECCS room cooler tubing identified by eddy current testing. Corrective actions included revising procedures to require periodic eddy current testing and replacing the associated cooling coil.

(c) The EPRI documents referred to in LRA Section B.2.6 for determination of allowable percent wall loss, plugging criteria, and projections for remaining life were used as guidance in the development of the procedure governing eddy current testing of heat exchangers. The specific documents are as follows:

- EPRI TR-106857, Volume 34, Preventive Maintenance Program Basis: Main Condensers, July 1988
- EPRI CS-5235, Recommended Practices for Operating and Maintaining Steam Surface Condensers, July 1987
- EPRI TR-100385, Balance-of-Plant Heat Exchanger Condition Assessment Guidelines, July 1992
- EPRI TR-101772, Electromagnetic NDE Guide for Balance-of-Plant Heat Exchangers, Rev. 2, December 1997
- ERPI TR-110392, Eddy Current Testing of Service Water Heat Exchangers for Engineers Guideline, Final Report, February 1999

The procedure governing eddy current testing contains criteria for establishing inspection timing, inspection interval reduction or expansion, and tube random sampling schemes based on criteria such as:

- Number of tubes plugged
- Rate of tube wall loss
- Evidence of tube cracking
- Wall degradation factors (e.g., flaw growth rate)

Similar criteria are provided in procedures governing other NDE methods (e.g., UT, radiography) utilized by the program.

Acceptance criteria for visual inspections may vary depending on a number of parameters associated with the particular heat exchanger inspected. However, in general, examples of the acceptance criteria to be used in determining the need for corrective actions are as follows:

- The number of plugged/ blocked tubes is less than that allowed as identified by review of heat exchanger load calculations or engineering judgment, as applicable.
- The evidence of ID tube scaling or slime formation on tube or head surfaces is less than that determined acceptable based upon review of historical inspection results and system operations and chemical treatment reliability/changes during the current operating period.
- The evidence of deep pitting (of a specified depth) underneath the nodules or general corrosion on other surfaces is less than that determined acceptable based upon review of historical inspection results. Corrosion in excess of the established amount results in: condition report generation, UT to determine remaining wall thickness, repair of locations (T_{min}) and those areas that may drop below T_{min} prior to the next inspection, if applicable.

Based upon historical inspection results, no evidence of divider/partition plate degradation or gasket extrusion has is present in the heat exchanger. In addition, review of system flows and D/Ps indicates no abnormalities. Excessive corrosion of the divider/partition plate in the seating area, plate warpage, or excessive gasket distortion/extrusion would require repair and engineering evaluation to estimate bypass flow.

Evaluations are performed for inspection results that do not satisfy the acceptance criteria for any of the above types of inspection and condition reports are initiated to document the concerns in accordance with the corrective action program. Resolution of the condition reports will include engineering evaluations, which assess remaining component life and determine the need for additional aging management activities.

(d) Cracking and loss of material results are documented in plant procedures. Evaluations are performed for inspection results that do not satisfy the acceptance criteria provided in the procedures and condition reports are initiated to document the concerns in accordance with the corrective action program. Resolution of the condition reports will include engineering evaluations, which assess remaining component life and determine the need for additional aging management activities.

In addition, eddy current testing is one of the inspection methods for the Heat Exchanger Test and Inspection Activities program. The procedure governing eddy current testing contains guidance required for prediction of remaining component life. This guidance is based on EPRI documents as identified in the response to (c) above.

(e) The EPRI/SANDIA reports identified in Aging Management Reference 3.1.2.15 of LRA Table 3.1-2 are EPRI 1003056, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Appendix G, Heat Exchangers," and Sandia National Laboratory Report SAND93-7070 UC-523, "Aging Management Guideline for Commercial Nuclear Power Plants – Heat Exchangers."

No Dresden-specific operating experience involving fouling of the isolation condenser tubing in the demineralized water environment was identified. However, EPRI 1003056, identifies fouling as an applicable aging effect for stainless steel tubing in treated water and primary water environments.