August 7, 2003

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

Dear Mr. Skolds:

By letter dated January 3, 2003, Exelon Generation Company, LLC (EGC) submitted, for the Nuclear Regulatory Commission's (NRC's) review, an application pursuant to 10 CFR Part 54, to renew the operating license for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. We are reviewing the information contained in the license renewal application (LRA) and have identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed request for additional information (RAIs) is from 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," and Appendix B, "Aging Management Programs."

We have provided these RAIs to Messrs. R. Stachniak and F. Polaski of your staff in parts between May 14 - June 13, 2003. The staff is willing to meet with EGC prior to the submittal of the responses to provide clarifications of the staff's RAIs.

Sincerely,

/**RA**/

Tae Kim, Senior Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-237, 50-249, 50-254,
and 50-265

Enclosure: As stated

cc w/enclosures: See next page

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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.

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.

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.

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.

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.

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.

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.

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.

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:

- 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.

- 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.

- 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.

RAI 3.1-10

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.

RAI 3.1-11

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.

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.

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.

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.

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.

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.

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?

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.

RAI 3.1-19

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.

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.

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

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

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

RAI 3.1-21

(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.

RAI 3.1-22

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.

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.

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.

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.

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."

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.

<u>RAI 4.2.3</u>

(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.

<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.

RAI 4.2.6

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.

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?

RAI 4.2-BWRVIPS

(a) 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.

(b) 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

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?

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.

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.

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)

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.

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.

RAI B.1.3

(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.

<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.

<u>RAI B.1.7</u>

(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."

<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 hte 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.

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.

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.

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 °F, 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.

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.

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.

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.

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.