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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

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 Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2

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 Patrick Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information for the Review of the License Renewal Applications for Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2," dated November 20, 2003

Exelon Generation Company, LLC (EGC) is submitting the additional information requested in email requests sent by Tae Kim (NRC) to EGC on October 23, 2003, and November 4 and 19, 2003. This additional information provides a response to questions regarding Sections 2.3, 2.4, 3.5, and associated Aging Management Programs sections of Reference 1 to support NRC review. In addition, EGC is revising the response to RAI 3.6-2 that was submitted in Reference 2.

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

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A098

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

Respectfully,

12/5/03
Executed on

Patrick R Simpson
Patrick R. Simpson
Manager – Licensing

Attachment : Response to Request for Additional Information – LRA Sections 2.3, 2.4, 3.5 and 3.6, and Associated Aging Management Programs

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Illinois Emergency Management Agency

Attachment

Response to Request for Additional Information

LRA Sections 2.3, 2.4, 3.5 and 3.6, and Associated Aging Management Programs

RAI 2.3.4.2-3 (Item 3.1.1.13) Supplemental Information Request

Based on the response to RAI 2.3.4.2-3, the staff understands that the Control Rod Drive (CRD) Return Line Nozzle has been capped, but not rerouted, and therefore augmented inspection for the nozzle is not required per NUREG-0619. The requirements in NUREG-0619 provide actions to be taken to address cracking in these nozzles. However, the aging effects of the cap and applicable weld is not covered in NUREG-0619. Therefore, the staff requests the following concerning the cap and weld which provides a pressure boundary function:

- Describe the configuration and location of the capped nozzle. This should include the existing base material for the nozzle, piping (if piping remnants exist) and cap material, any welds and material type (i.e. 82/182).
- Describe how this weld and cap is managed (i.e. BWRVIP-75).
- Discuss how the event at Pilgrim (leaking weld at capped nozzle) may or may not apply to Dresden and Quad Cities. Include in your discussion the past inspection techniques applied, the results obtained, mitigative strategies, and weld repairs, etc.

Response

- At Dresden, the current configuration includes 3" stainless steel cap welded to a new stainless steel safe-end, welded to the original carbon steel nozzle. Also, a ½" sockolet is welded to the safe-end going to a capped spare ¾" stainless steel line on Dresden Unit 3 only.

  At Quad Cities, the current configuration includes a new 3" stainless steel cap welded to a new stainless steel safe-end, welded to a new 1 ¼" long carbon steel pup piece (pipe) (with a stainless Steel overlay) welded to the original carbon steel nozzle.

- The aging management for this section includes ASME Section XI for the nozzle as stated in Aging Management Program B.1.6, and ASME Section XI Aging Management Program B.1.1 for the remaining portion (safe-end, cap, and welds).

- The October 1, 2003 event at Pilgrim does not apply to Dresden and Quad Cities based on the following differences:

  - Pilgrim welded their cap directly to the nozzle. Dresden and Quad Cities have installed a new safe-end between the nozzle and cap.
  - The Pilgrim cap was Alloy 600. The Dresden Safe-ends are 316L and the Caps are 304L. The Quad Cities Cap and safe-end are 316L.
  - Pilgrim used inconel 82/182 alloy weld filler material. Dresden and Quad Cities used E308L. Quad Cities also used E309L for the dissimilar metal weld.
  - Pilgrim had initial weld deficiencies (lack of fusion) that required weld repair. The Dresden and Quad Cities welds were completed without incident (no recordable indications).
  - Pilgrim installed the cap in 1977. Subsequent to the Pilgrim installation it was determined that Inconel 600 caps and inconel 82/182 nozzle to cap butt welds were, under specific conditions, susceptible to stress corrosion

cracking. Dresden installed the caps in 1993 and1986 (Units 2 and 3 respectively) and Quad Cities installed the caps in 1989 and 1990 (Units 1 and 2 respectively) and considered this new operating experience into account in the design of the modification.

- NDE completed since the replacement of the nozzles and caps has included Radiographic and penetrant testing (initial installation) and subsequent ultrasonic and penetrant testing per the ISI program. No reportable indications have been identified.

- The nozzle-to-safe end weld is a category B-F weld, and the safe end-to-cap weld is a category B-J. As such, all of the welds listed below were removed from the scope of BWRVIP-75 and GL 88-01. These welds are included in the Risk Informed ISI program.

- Additionally, the response to RAI 3.1-9 did not include these capped lines. They were omitted from the list as they are not installed piping lines. Therefore, the table below amends the response to RAI 3.1-9. These capped lines have been included in Aging Management Program B.1.1 "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

Unit	**System**	**Line No.**	**Material**	**Weld Type**	**Drawing (Coordinates) and Comments**
D-2	Control Rod Drive	Capped Return Line	Stainless Steel	Butt	LR-DRE-M-26-1 (E-6)
D-3	Control Rod Drive	Capped Return Line	Stainless Steel	Butt And socket	LR-DRE-M-357-1 (B-4)
Q-1	Control Rod Drive	Capped Return Line	Stainless Steel	Butt	LR-QDC-M-35-1 (G-5)
Q-2	Control Rod Drive	Capped Return Line	Stainless Steel	Butt	LR-QDC-M-77-1 (G-5)

RAI 2.4-2 Supplemental Information Request

In its response, the applicant identified a specific component group for all items listed in the RAI, except for the Double Gasket. Since the double gasket is on a regular replacement schedule, the staff concurs that it does not require aging management for license renewal.

Several of the component group designations, however, appear to be incorrect. The response to RAI 2.4-2 lists at least three items that appear to be Class MC supports (items c, d, and j), but the LRA Table number and component group referenced for each item leads to the Structures Monitoring Program, not IWF. The response to RAI 2.4-2 also lists a number of items that appear to be Class 1 supports (items a, b, and f (regarding anchor bolts)), but the LRA Table number and component group referenced for each item leads to the Structures Monitoring Program, not IWF.

Response

a) Reactor Vessel to Biological Shield Stabilizers (D-UFSAR Figs. 3.9-1 and 2): These supports receive a VT-3.

The correct component group designation for this item is found in Table 2.4-15 Component Supports. These are included in the Component Group named "Support Members (Includes Spring Hangers)". The correct AMR reference for these components is AMR Ref. 3.5.1.31 (IWF).

b) Biological Shield to Containment Stabilizer (D-UFSAR Figs. 3.9-1 and 2 and QC-UFSAR Fig. 3.9-5 and 8) – These supports are not currently inspected. However, prior to the end of the current term of operation the IWF program will be augmented to cover these Class MC supports, requiring a VT-3 of the accessible areas.

The correct component group designation for this item is found in Table 2.4-15, Component Supports. These are included in the Component Group named "Support Members (Includes Spring Hangers)". The correct AMR reference for these components is AMR Ref. 3.5.1.31 (IWF).

c) RPV Male Stabilizer Attached to Outside of Drywell Shell (QC-UFSAR Figs. 3.9-5 and 8) – This is a subset of same support discussed in item b above.

d) RPV Female Stabilizer and Anchor Rods (also referred to as Gib) embedded in Reactor Building concrete wall (D-UFSAR Fig. 3.9-1 and QC-UFSAR Figs. 3.9-8 and 9) – This is a subset of same support discussed in item b above.

f) Reactor Vessel Support Skirt and Anchor bolts (D-UFSAR Figs. 3.9-2 and 3 and QC-UFSAR Figs. 3.9-5, 6 and 10) – The integral attachment to the reactor vessel receives a surface (magnetic particle or liquid dye penetrant) examination as part of the ISI program, the rest of the support receives a VT-3 examination as part of the IWF program. Aging management reference 3.1.2.33 should read as follows.

Ref No	Component Group	Material	Environment	Aging Effect	Aging Management Program	Discussion
3.1.2.33	Support Skirts and Attachment Welds	SA 302Gr B Welds Low Alloy Steel	Containment Nitrogen	Crack initiation and growth/ Cyclic loading; Loss of material/ Environmental corrosion (i.e. pitting corrosion, general corrosion, etc.)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1); ASME Section XI, Subsection IWF (B.1.27)	NUREG-1801 does not address crack initiation, growth/cyclic loading or environmental corrosion for support skirts and attachment welds.

j) Drywell steel support skirt and anchor bolts (QC-UFSAR Figs. 3.9-5 and 7) –The steel support member is part of the class MC support, however, it is encased in concrete and not accessible for and is exempt from examination per ASME Section XI, IWF-1230 (components encased in concrete).

The correct component group designation for this item is found in Table 2.4-15, Component Supports. These are included in the Component Group named "Support Members (Includes Spring Hangers)". The correct AMR reference for these components is AMR Ref. 3.5.1.31 (IWF).

RAI 2.4-7 Supplemental Information Request

The additional information provided by the applicant in its RAI response sufficiently answers the questions posed by the staff, with one (1) exception. The staff cannot determine whether the applicant has appropriately addressed the Quad Cities intake flume/canal in its scoping and screening review. As stated in the RAI response, "The intake flume boundaries includes the topographic basin from the high point (at approximately 565' elevation) on the river bottom between the crib house and the main river channel on the west side and extending to the crib house on the east side. This basin is rock and earthen bottom. LRA Table 2.4-11, Component Group Concrete Walls, addresses the crib house walls." The applicant has not indicated whether the intake flume/canal is within the license renewal scope. If it is, where in the LRA is the AMR for the basin? If the intake flume/canal has not been included in the license renewal scope, then the applicant needs to provide its technical basis for that determination. This is Open Item 2.4.11.2-1.

Response:

The Quad Cities intake flume is in the scope of license renewal. The addition of the line item "Earthen Structures" to LRA Table 2.3.3-22 in the response to RAI 2.4-7 was done incorrectly in that Exelon designated "Earthen Structures" as being for Dresden only. This is not correct. The line item "Earthen Structures" in LRA Table 2.3.3-22 is for both Dresden and Quad Cities. The "Dresden Only" entry supplied in the original response to RAI 2.4-7 will be deleted. Aging management of the Quad Cities intake flume/canal is per aging management reference 3.5.1.22 in Table 3.5-1, and specifies aging management program B.1.31, RG 1.1.27, Inspection of Water-Control Structures Associated with Nuclear Power Plants.

RAI 2.4-9 Supplemental Information Request

The applicant also clarified the aging management review of cranes and hoists in its response to RAI 2.4-9. Cranes and hoists related to refueling are included under Auxiliary Systems, while all other cranes and hoists within the scope of license renewal are included under Structures. In all cases, the aging management program "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems", described in LRA Section B.1.15, is credited to manage loss of material due to general corrosion and wear. The staff reviewed this AMP to ensure that all cranes and hoists are included in its scope. The AMP description in LRA Section B.1.15 only addresses load handling systems related to refueling.

The applicant needs to define an enhancement to the scope of the AMP, to include inspection of all cranes and hoists within the scope of license renewal.

Response

The AMP in LRA Section B.1.15 is titled identically to GALL program XI.M23. The title is misleading since in both cases, GALL and the LRA, the program covers all cranes and hoists within the scope of license renewal, not just those related to refueling. The program description for B.1.15 already includes "bridge and trolley structural components for systems within the scope of 10 CFR 54.4 and other load handling systems within the scope of license renewal." Therefore the program scope is not restricted to cranes and hoists related to refueling.

RAI 3.5-4 Supplemental Information Request

The staff finds that appropriate design provisions to ensure that concrete does not exceed prescribed ACI code limits are identified in the response. However, the statement "The Dresden and Quad Cities Groups 1-5 concrete structures were installed in accordance with ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, Appendix A."appears to be incorrect, given the dates of first commercial operation for these units. On this basis, the staff does not accept the applicant's response as submitted. The applicant needs to identify the correct code of record and the temperature limits prescribed in that code; to re-state the design provisions implemented at Dresden and Quad Cities to ensure satisfaction of the prescribed limits; and to describe plant-specific operating experience related to concrete exposure to elevated temperature, for all four (4) units. This is Open Item 3.5.2.2.1.3-1.

Response

Dresden and Quad Cities Groups 1-5 structures were designed to ACI 318-63, Building Code Requirements for Reinforced Concrete. This standard does not address susceptibility of concrete to aging effects associated with elevated temperatures. However, Appendix A of ACI 349-85 is specifically cited by NUREG-1801 and EPRI Report 1002950, "Aging Effects for Structural Components (Structural Tools), Revision 1," August 2003, as providing the temperature criteria (< 150°F general, < 200°F localized) to be used in determining the susceptibility of concrete to aging effects associated with elevated temperatures. These criteria were used to determine the susceptibility of the Dresden and Quad Cities Groups 1-5 structures to the subject aging effect (reduction of strength and modulus due to elevated temperature). Since these structures are not exposed to general temperatures in excess of 150°F or localized temperatures in excess of 200°F, it was determined that the aging effect was not applicable.

RAI 3.5-6 Supplemental Information Request

Based on the applicant's response to part (a) of this RAI, there are a total of 120 bellows within the scope of license renewal (32 for each Dresden unit; 28 for each Quad Cities unit). Of the 120 total, 24 bellows have been identified as degraded due to TGSCC over the period September 1990 through January 2003, and have been replaced (23) or taken out of service (1). The applicant states in part (d) of its response that "Degraded bellows assemblies identified since 1991 were identified utilizing the methodology developed to comply with the exemptions."

Since there are 96 original bellows still in place, and the period of extended operation will begin in approximately 10 years, it is not clear to the staff that reliance on Appendix J Leak Rate Testing and IWE Examination Category E-A to manage aging for license renewal is sufficient, without an additional commitment to continue the testing methodology described in (1) through (6) under part (d) of the RAI response. The applicant needs to specifically credit this testing methodology for aging management of bellows during the period of extended operation. This is Open Item 3.5.2.2.1.7-1.

Response

The testing methodology summarized in steps (1) through (6) under part (d) of the RAI 3.5-6 response is a summary of the testing methodology detailed in the NRC February 6, 1992 letter from Bruce A. Bolger to Thomas J. Kovach, granting the exemption from the testing requirements of Appendix J to 10 CFR Part 50 for Dresden and Quad Cities Nuclear Power Stations, and as detailed in the NRC February 9, 1995 letter from Robert M. Pulsifer to D. L. Farrar granting a revision to the exemption. As such, in accordance with the conditions of the exemption, Exelon will utilize this testing methodology for each non-testable two-ply bellows assembly (original design). As stated in the NRC letter granting the exemption, "Upon replacement with a testable bellows assembly, that bellows will no longer be included in the Exemption and will be required to be tested in accordance with the normal Type B program. Similarly, if a method is developed which insures a valid Type B test on one or more bellows assemblies, those bellows will also be excluded from the Exemption and will be required to be tested in accordance with the normal Type B test program."

Therefore, to the extent that any non-testable two-ply bellows assembly (original design) remain during the period of extended operation, they will continue to be tested utilizing the methodology summarized in steps in (1) through (6) under part (d) of the RAI 3.5-6 response.

<u>RAI 3.5-11 Supplemental Information Request</u>

Since underwater accessible areas will be inspected, any occurrences of abrasion erosion or cavitation will be detected in these areas. However, the staff is unclear about the applicant's justification that abrasion erosion and cavitation do not require aging management for inaccessible areas. Part of the definition of "inaccessible areas", in part (1) of the response, is "where...high flow rates make diver entrance unsafe without a dual unit outage." These would appear to the areas most susceptible to abrasion erosion and possibly cavitation. The applicant needs to quantify "high flow rates". In addition, it is unlikely that the water velocity is a uniform 3.68 fps across the entire flow area in the intake tunnel, adjacent to the circulating water pump. The applicant needs to consider a realistic velocity profile in estimating the maximum water velocity. This is Open Item 3.5.2.4.2.2-1.

Response

The inaccessible areas are better described as those areas where continuous flow makes diver entrance unsafe without a dual unit outage. In other words, the common areas in the crib house intake outside of the individual bays to the circulating water pumps are those considered inaccessible. The highest velocities experienced in the underwater structures will be in the individual circulating water bays, adjacent to the circulating water pumps themselves. The intake for each of the circulating water bays is a tapering volume which at the largest point is 25'-4" wide by 21'-6" high at normal water level. At the opening to the circulating water pump suction area the bay measures 25'-4" wide by 8' high. The 3.68 fps velocity reported in the original response to RAI 3.5-11 corresponds to a cross sectional area of 12' wide by 8' high, which are the dimensions of the tapered intake tunnel (bay) at the pump suction centerline. Therefore the flow velocities in most of the individual circulating water pump bay area and in the inaccessible crib house common areas outside of the traveling screens are appreciably smaller.

The individual circulating water pump bays are accessible and will be inspected since they can be taken out of service during the applicable unit outages. The inspection scope covers all of the tapered area described above. Since the limiting locations for flow velocities and potential erosion effects are to be inspected, the inaccessibility of the areas of lower flow velocity is not detrimental to aging management.

RAI 3.5-12 Supplemental Information Request

The applicant has not specifically addressed a key element of this RAI. The ultimate heat sink raw water is considered aggressive by its nature, and all concrete exposed to it needs to be managed for these aging effects/mechanisms. The applicant was asked to specifically discuss whether.... below-water concrete in water-control structures is being excluded from aging management, and if applicable, submit a detailed technical justification for not managing aging of ......below-water concrete in water-control structures, in light of past industry operating experience indicating there is a significant potential for degradation. The applicant needs to submit its aging management review for concrete exposed to the ultimate heat sink raw water, and either identify the credited aging management programs, or submit a detailed technical justification for not managing aging of concrete exposed to raw water. This is Open Item 3.5.2.4.2.2-2.

Response

All in-scope below-water concrete (submerged) exposed to the ultimate heat sink raw water environment will be managed for aging except the inaccessible common area in the crib house intake outside of the individual bays to the circulating water pumps where continuous flow makes diver entrance unsafe without dual unit outage (see response to RAI 3.5-11 supplemental information request). Aging Management Program B.1.31, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, provides for managing the aging effects of in-scope accessible concrete exposed to ultimate heat sink raw water environment at Dresden and Quad Cities.

The in-scope concrete exposed to the ultimate heat sink raw water environment at Quad Cities is addressed in LRA Section 2.4.11. In Table 2.4-11, the Component line items "Concrete Canal Weirs (Quad Cities Only)," "Concrete Walls" and "Concrete Slabs", each with the component intended function of "heat sink" includes Discharge flumes and the crib house below-water concrete structures.

Similarly, the in-scope concrete exposed to the ultimate heat sink raw water environment at Dresden is addressed in LRA Sections 2.4.11 and 2.3.3.22. In Table 2.4-11, the

Component line items "Concrete Walls" and "Concrete Slabs", each with the component intended function of "heat sink" includes the crib house below-water concrete structures.

The discharge outfall concrete structure at Dresden has been added to LRA Table 2.3.3-22 under Component Group – Concrete Slabs and Concrete Walls as stated in the original response to RAI 2.4-7. Aging Management Reference 3.5.1.22 discusses the aging management of the discharge outfall concrete components submerged in a raw water environment.

RAI 3.5-13 Supplemental Information Request

The staff finds that the applicant's response related to ASTM A193, Grade B7 bolting material is sufficient to establish that the upper limit on yield strength is < 150 ksi; consequently, the additional inspections of XI.M.18 Bolting Integrity are not warranted for A193, Grade B7. However, in part (c) of its response, the applicant has assumed that the bolt material used in "a friction type connection at the reactor skirt base" is ASTM A193, Grade B7 or equivalent.

The applicant needs to (1) provide a definitive basis to support a determination that the yield strength is < 150 ksi, or (2) commit to inspection in accordance with XI.M.18 Bolting Integrity. This is Open Item 3.5.2.4.5.2-1.

Response

The subject bolts at the reactor skirt base were installed in accordance with General Electric Drawing 158B7707, Reactor Vessel Support Bolting. This drawing identifies the bolt material and diameter as ASTM A490 and 2" respectively. ASTM A490 identifies a yield strength for this material of 130 ksi. However, the ASTM specifies a maximum Brinell Hardness number of 352 HB for subject bolting. Based on ASTM A370, Standard Specification, Section 1, Volume 01.01, a maximum Brinell Hardness number of 352 HB (interpolated between 344 HB and 353 HB) equates to a tensile strength of 170.4 ksi (interpolated between 166 ksi and 171 ksi). Therefore, the maximum yield strength of the subject bolting is rounded to an approximate tensile strength threshold value of 170 ksi value.

Since the yield strength for this material cannot be confirmed to be less than 150 ksi, Exelon will commit to inspection of the subject bolts in accordance with NUREG-1801, Program XI.M18, Bolting Integrity. Per Program XI.M18 requirements, the inspections will consist of VT-1 examinations of the surface of the bolts, and nuts. These inspections will be performed at a frequency not to exceed every ten years.

LRA Sections B.1.12, Bolting Integrity and A.1.12, Bolting Integrity (for each site) will be revised as necessary to include this new inspection requirement.

RAI 3.5-15 Supplemental Information Request

The staff finds that the applicant's response clarifies the aging management review and the credited aging management programs for the clevis pins. The applicant has identified cracking due to SCC as an applicable aging effect/aging mechanism for stainless steel clevis pins submerged in torus grade water, and credits the Water

Chemistry program for aging management. The staff position is that some verification of the effectiveness of the Water Chemistry program is necessary, and has previously accepted a one-time inspection as the verification method. This is consistent with the applicant's approach to aging management for all other aging effects applicable to the clevis pins submerged in torus grade water. The applicant needs to describe its methodology to verify the effectiveness of the Water Chemistry program in preventing cracking due to SCC for stainless steel clevis pins submerged in torus grade water. This is Open Item 3.5.2.4.5.2-2.

Response

The normal maximum operating water temperature in the torus is 95 deg-F. The typical flow conditions are either low flow or stagnant flow. With process temperatures below 140 deg-F, EPRI TR 1003056, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools", Appendix A states that cracking due to SSC is very unlikely to occur. In addition, Exelon believes that water chemistry controls sufficient to prevent loss of material due to pitting and crevice corrosion in the torus are also sufficient to prevent stress corrosion cracking in that system. Nonetheless, Exelon will include inspection for stress corrosion cracking as part of its one-time inspection to validate the effectiveness of the Water Chemistry Program (LRA Appendix B.1.2) in managing the aging of stainless steel components in the torus.

<u>RAI 3.5-16 Supplemental Information Request</u>

The staff finds the applicant's AMR and selection of the AMP for thermowells installed in the suppression chamber shell to be acceptable, based on consistency with the guidance in GALL for similar material and environment. Based on the information provided, the staff accepts the applicant's conclusion that the thermowells are not susceptible to SCC. This part of RAI 3.5-16 is resolved.

The staff accepts the applicant's use of IWF to manage loss of material for the stainless steel pipe support stanchions used on the recirculation piping 28" lines at Dresden and Quad Cities. This is consistent with GALL. However, the applicant has not addressed the potential for cracking due to SCC. To enable the staff to complete its evaluation, the applicant is requested to provide its aging management review for cracking due to SCC for the stainless steel pipe support stanchions used on the recirculation piping 28" lines at Dresden and Quad Cities. This is Open Item 3.5.2.4.5.2-3.

Response

Stress corrosion cracking for stainless steel is discussed in EPRI 1003056 "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," Appendix E, Section 3.2.2, and EPRI report TR-114881, "Aging Effects for Structure and Structural Components," Section 2.3.2.2. These EPRI documents were used to evaluate the aging of the Class 1 stainless steel supports since stainless steel supports are not evaluated in GALL. The aging mechanisms/effects associated with the material and environment combination for these supports apply to all pipe supports regardless of pipe code classification. For this reason, the aging mechanism/effect of stress corrosion cracking obtained from EPRI 1003056 "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools" applies to the subject Class 1 supports.

Stress corrosion cracking is a mechanism requiring a tensile stress, a corrosive environment, and a susceptible material in order to occur (Ref. EPRI 1003056, App. E, Section 3.2.2, Figure 1 and EPRI TR-114881, Sec 2.3.2.2).

The stainless steel pipe support stanchions used on the recirculation piping 28" lines at Dresden and Quad Cities are located inside the drywell containment. The drywell environment is not conducive to stress corrosion cracking, since it is inerted (<4% oxygen) and has a maximum relative humidity of 90%. Since these supports do not experience any corrosive environment, the stress corrosion cracking aging mechanism does not exist for these supports. As such, these supports do not require aging management for SCC.

RAI 3.5-17 Supplemental Information Request

The staff has identified the need for additional information, and has also identified a discrepancy between the responses to RAI 3.5-15 and RAI 3.5-17:

(1) According to the applicant's response to RAI 3.5-15, "The line item for Support Members in Table 2.4-15 incorrectly referenced 3.2.2.79, 3.2.2.80, and 3.2.2.81. These references should have been designated as aging management references for Support References. Aging Management References 3.5.1.29, 3.5.1.31, and 3.5.2.14 are correct. New Aging Management References should have included 3.5.2.17, 3.5.2.18, and 3.5.2.19, as shown below in Table 3.5-2." In the RAI 3.5-17 response, part (a), there is no indication of the correction described in the RAI 3.5-15 response. The applicant needs to clarify this.

(2) Part (b) of the response to RAI 3.5-17 is not acceptable. The supports in question are not Class MC supports. The systems involved are most likely Class 2. In addition, the reference to "(components that are part of the reactor coolant pressure boundary)" appears to be misplaced. The applicant needs to re-submit its justification for not crediting IWF.

(3) Using only carbon steel HPCI torus suction check valves as the basis for the one-time inspection does not address the potential aging effects for stainless steel support members. The applicant needs to describe how the one-time inspection will address aging effects for stainless steel support members.

(4) As previously stated in the evaluation of the response to RAI 3.5-15, the staff position is that some verification of the effectiveness of the Water Chemistry program is necessary, and has previously accepted a one-time inspection as the verification method. This is consistent with the applicant's approach to aging management for all aging effects applicable to the submerged supports, except for cracking due to SCC. The applicant needs to describe its methodology to verify the effectiveness of the Water Chemistry program in preventing cracking due to SCC for stainless steel support members submerged in torus grade water. This is Open Item 3.5.2.4.5.2-4.

Response

1) In preparing its original response to RAI 3.5-17, Exelon failed to coordinate the discussion appropriately with the LRA changes described in response to RAI 3.5-15. The first paragraph in response to RAI 3.5-17 (1) is revised to read as follows.

LRA Aging Management References 3.2.2.79, 3.2.2.80, and 3.2.2.81 have been replaced by references 3.5.2.17, 3.5.2.18 and 3.5.2.19 (See RAI 3.5-15). These Aging Management References are applicable for support members submerged in a torus water environment. Support members in a torus water environment include submerged supports for Low Pressure Coolant Injection (LPCI) System (Dresden, only) piping, Residual Heat Removal (RHR) System (Quad Cities, only) piping, High Pressure Coolant Injection (HPCI) System piping, Reactor Core Isolation Cooling (RCIC) System (Quad Cities, only) piping, and Main Steam System (relief valve tailpipe) piping.

2) As revised in the original response to RAI 3.5-15, IWF is credited in Aging Management References 3.5.1.31 and 3.5.2.14, which are applicable for structural members with a Component Intended Function of "structural support"; and IWF is credited in Aging Management References 3.5.1.31, 3.5.2.5 and 3.5.2.23, which are applicable for clevis pins.

3) Exelon believes that Torus Water Chemistry controls sufficient to prevent the aging effects of loss of material due to general, crevice and pitting corrosion in the carbon steel HPCI torus suction check valves will also be sufficient to prevent aging effects in stainless steel support members and components. Nevertheless, Exelon will provide a one-time inspection of selected stainless steel clevis pins in the submerged (torus grade water) environment to confirm the effectiveness of Torus Water Chemistry controls in preventing the aging effect/ mechanism of cracking/ stress corrosion cracking. Where the selected stainless steel clevis pins interface with uncoated carbon steel support members, the interfacing support members will also be inspected for the aging effect/ mechanism of loss of material/ galvanic corrosion.

4) As described in the preceding response to RAI 3.5-17, question # 3 and in RAI 3.5-15 Supplemental response, Exelon will provide a one-time inspection of selected stainless steel clevis pins in the submerged (torus grade water) environment to confirm the effectiveness of Torus Water Chemistry controls in preventing the aging effect/ mechanism of cracking/ stress corrosion cracking. Where the selected stainless steel clevis pins interface with uncoated carbon steel support members, the interfacing support members will also be inspected for the aging effect/ mechanism of loss of material/ galvanic corrosion.

RAI 3.6-2 Supplemental Information Request

In response to RAI 3.6-2, at Quad Cities all but three electrical penetrations are part of the station EQ program. These three penetrations serve circuits (such as drywell booster fans and main steam line vibration monitoring instrumentation) that do not perform any electrical intended function. The staff is concerned about a leak in penetration due to electrical fault on these circuits. Please provide details about these circuits (i.e., energized during shut down only and power supply is disconnected during plant operation, etc.). Discuss why the aging of the insulation do not have any effect on the penetration damage curve so that penetration seal integrity is maintained as a part of containment pressure boundary.

Response

The three electrical penetrations at Quad Cities that are not part of the station EQ program are included within the scope of license renewal. However, they only perform a pressure boundary function for the primary containment and do not have any electrical related intended functions. The pressure boundary function for these penetrations is managed under ASME Section XI, Subsection IWE (B.1.26) and 10 CFR Part 50, Appendix J (B.1.28).

Penetration #	Load
1-X102B	Drywell 1 Vent Booster Fan (Ref 4E-1670G) Junction Box IRB-262 Vibration Instrumentation
2-X100A	Vibration Instrumentation Vibration Instrumentation
2-X105A	Drywell 2 Vent Booster Fan (Ref 4E-2670H) Junction Box IRB-180 Vibration Instrumentation

The Drywell Vent Booster Fans are continuously energized during plant operations. The circuit for these fans is protected by redundant 100 amp in-scope circuit breakers. The cables from the MCC to the penetrations and from the penetrations to the fans are in-scope and managed by aging management program B.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements. The Conax penetration feed-through-modules are # 2 AWG solid copper conductors insulated with polyimide film. The circuits are designed such that the 100 amp breakers are coordinated to clear all fault currents before the short circuit capacity of the # 2 AWG feed-through-modules is exceeded thus preventing damage to the penetration seal integrity. There are no credible aging effects that reduce the short circuit capacity of solid copper conductors. Short circuit capacity is based on the circular mills of the copper conductor.

The vibration Instrumentation circuits are low voltage, milliamp circuits protected by fuses. Fault currents are in the milliamp range and not severe enough to cause damage to the # 18 AWG feed-through-modules. The cables for these instrumentation circuits are in-scope and managed by aging management program B.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

The design of the Conax penetration module is a stainless steel tube that is sealed at both ends with polysulfone. Solid copper polyimide film insulated conductors passes though the stainless steel tube and are molded into the polysulfone seal at both ends to provide a leak proof seal. A visual inspection of the exposed polyimide film insulation will not provide any indication of the leak tightness of the penetration because the insulation cannot be visually inspected once it passes into the polysulfone seal. The aging management programs that are used to manage the aging of the pressure boundary function are Containment ISI (B.1.26) and Containment leak rate test (B.1.28).

Identical Conax EQ penetrations are installed at the Dresden station. The Dresden Conax EQ penetrations are qualified for 60 years of normal and one-year accident/post accident conditions in accordance with IEEE 323-1983 requirements and NUREG-0588, Category I. The Quad Cities Conax penetrations are bounded by the environmental qualification reports approved for Dresden.

In summary,

a) Electrical faults are mitigated by the circuit protection devices prior to damaging the feed through conductor or insulation.
b) Visual inspection of the pigtail insulation provides no indication of the integrity of the seal.
c) These penetration do not perform an electrical intended function that supports 10 CFR 54.4 (a) (1) (i), (ii), (iii) (2) or (3).
d) The pressure boundary function is managed by aging management programs Containment ISI (B.1.26) and Containment leak rate test (B.1.28).
e) Identical Conax EQ penetrations are installed at Dresden and are qualified for 60 years.

Therefore, using aging management programs Containment ISI (B.1.26) and Containment leak rate test (B.1.28) to manage the aging effects of the penetrations provides reasonable issuance that the License Renewal intended function of the Quad Cities Non-EQ penetrations will be maintained during the prior of extended operation.

<u>RAI B.1.30 Supplemental Information Request</u>

The additional information provided by the applicant in its response to RAI B.1.30 sufficiently answers the questions posed by the staff, with (2) exceptions. It is not clear whether the category "Piping Component Supports including immediately adjacent piping/tubing," listed in the response to item (a) of the RAI is meant to include non-ASME piping supports. It is also not clear as to why the Structures Monitoring Program does not include "standard components such as snubbers, struts and spring cans." In order to completely resolve the response to this RAI, the staff requests that the applicant confirm that:

1. the B.1.30 program covers non-ASME piping supports, and
2. there are no snubbers, struts and spring cans on non-ASME piping and components.

This is Confirmatory Item 3.0.3.14.2-1.

Response:

Exelon has reviewed the supplemental Information Request and provides the following clarification and confirmation.

1) The Structure Monitoring Program, B.1.30, includes non-ASME piping supports for aging management. The selection of component supports includes a representation of supports throughout the plant, considering environmental conditions as well as configuration.

2) There are standard components such as snubbers, struts, and spring cans on non-ASME piping and components that are in-scope of the License Renewal, which are required to be managed for aging. The Structural Monitoring

Program, B.1.30, will inspect the non-ASME component supports including the standard components. The in-scope non-ASME component supports are addressed in LRA Section 2.4.15, Table 2.4-15 under the Component Groups "Support Members" with a "Non-S/R Structural Support" component intended function. Aging Management Reference 3.5.1.29 discusses the aging management of the non-ASME component supports.

RAI B.1.31 Supplemental Information Request

The additional information provided by the applicant in its response to RAI B.1.31 did not completely address all of the staff's request for information and raised some additional concerns as discussed below. The applicant is requested to provide the following information:

1. The response to RAI B.1.31(a) states the parameters monitored for the concrete structures (Dresden Unit 1 and 2/3 Crib House and Quad Cities Unit 1 / 2 Crib House) included in the existing B.1.31 program. However, the RAI response also indicates that the Dresden intake and discharge canals are currently monitored under this existing program. Based on the information provided in the response to RAI 2.4-7, the staff understands that these canals are earthen structures. Therefore, the applicant is requested to explain what parameters are monitored for the earthen structures under this existing program.

2. The response to RAI B.1.31(a) did not explain how the condition of water control structures within the scope of license renewal that are not included in the existing program are currently monitored at Dresden and Quad Cities. The applicant is requested to provide this information for all structures and components identified in the response to RAI 2.4-7 as being within the scope of license renewal, as well as any other applicable structures and components that may not have been listed by the staff as part of RAI 2.4-7.

3. The response to RAI B.1.31(b) only described the operating experience with regard to the Dresden intake and discharge canals and the Dresden cooling lake (which is stated as being out of the scope of license renewal). The applicant is requested to describe the operating experience with regard to the inspection of all essential structural elements of the ultimate heat sink for both Dresden and Quad Cities as identified in the response to RAI 2.4-7.

4. The response to RAI B.1.31(d) does not discuss any existing procedures or planned enhancements related to the inspection of earthen structures. The applicant is requested to describe these procedures since it is clear that earthen structures are being monitored under the B.1.31 program.

5. The response to RAI B.1.31 does not address the Quad Cities intake flume/canal. The response to RAI 2.4-7 discusses the intake flume/canal boundaries, but does not specify whether the flume/canal is included in the license renewal scope, and does not provide a reference for the aging management review of the topographic basin. The staff requests the applicant to clarify whether the Quad Cities flume/canal, including the topographic basin, is monitored under the B.1.31 program. If it is, describe the monitoring procedures used. If it is not, explain the technical basis for its exclusion.

6. For the structures and components of the ultimate heat sink that are not currently being inspected under an existing program, the staff requests the applicant to provide a commitment to perform a baseline inspection of typical portions of each structure or component prior to the period of extended operation, to identify and correct any problems affecting performance of intended functions.

7. The staff notes that in LRA Section A.1.31 for both Dresden and Quad Cities there is no mention of earthen structures in the description of the RG 1.127 program for the UFSAR Supplement. The applicant is requested to revise these supplements to specifically identify earthen structures as being within the scope of this program, and also to include a discussion of any other significant changes in the scope of this program that have occurred as a result of the applicant's responses to the staff RAIs related to this program.

This is Open Item 3.5.2.3.4.2-1.

Response:

(1) The canals are earthen structures that require aging management. Specifically, these structures are vulnerable to the buildup of sedimentation. Existing station procedures monitor the aging effect, "Loss of Form due to Sedimentation", to ensure that the required volume of water is available in the ultimate heat sink to support emergency cooling conditions. This includes the forebay at Dresden and the forebay at Quad Cities. (Note the "Dresden Only" annotation in table 2.3.3-22 for Component Group "Earthen Structures" provided in RAI 2.4-7 was incorrect and has been deleted.)

The Quad Cities Earthen Structure consists of a bay excavated from the river front area down and into the existing bedrock up to the in-scope concrete structures. The sidewalls of this structure are engineering designed earthen slopes and are covered with rip-rap, both above and below the water line.

The Dresden Earthen Structure was excavated through the soil. The actual canals are excavated from bedrock. The soil portions above the canals are capped with concrete.

The aging management review of these structures found that "Loss of Form" was the only applicable aging effect, and sedimentation the only applicable aging mechanism contributing to this effect based on the design and configuration of these structures.

(2) The existing Aging Management activities for the in-scope Quad Cities components as discussed in RAI 2.4-7 are:

- Intake Flume – Aging management of the earthen portion of this structure is discussed in the response to (1) above. Concrete portions are addressed in LRA Table 2.4-11, under Component Group, "Concrete Walls", and Aging Management Reference 3.5.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31.

- 16' diameter discharge piping – This piping is addressed in LRA Table 2.3.3-22, under Component Group, "Piping and Fittings", and Aging Management Reference 3.3.1.15, Open Cycle Cooling Water Program B.1.13.

- 96" Ice Melting Line, including Gate – The ice melt line is addressed in LRA Table 2.3.3-22, under Component Group, "Piping and Fittings", and Aging Management Reference 3.3.1.15, Open Cycle Cooling Water Program B.1.13. The gate is addressed in LRA Table 2.3.3-22, under Component Group, "Valves" and Aging Management Reference 3.3.2.278, Open Cycle Cooling Water Program B.1.13 and Aging Management Reference 3.3.2.300, Bolting Integrity Program B.1.12.

- Discharge Flume/Canal – This structure is addressed in LRA Table 2.4-11, under Component Group, "Concrete Walls", and Aging Management Reference 3.5.1.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31. However, these aging management activities are an enhancement and are not currently implemented.

- Weir Gate in discharge canal – This component is addressed in LRA Table 2.4-11, under Component Group, "Concrete Walls", and Aging Management Reference 3.5.1.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31. However, these aging management activities are an enhancement and are not currently implemented.

The existing Aging Management activities for the in-scope Dresden components as discussed in RAI 2.4-7 are:

- Intake flume/canal – Aging management of the earthen portion of this structure is discussed in the response to (1) above. Concrete portions are addressed in LRA Table 2.4-11, under Component Group, "Concrete Walls", and Aging Management Reference 3.5.1.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31.

- Crib House Stop Logs – These components are addressed in LRA Table 2.3.3-22, Component Group "Stop Logs" and Aging Management Reference 3.3.2.304, with no aging management required (as supplied in RAI 2.4-7 response).

- Crib house dewatering valves and trash rake refuse pit – The valves are addressed in LRA Table 2.3.3-22, under Component group, "Valves", and Aging Management Reference 3.3.2.278, Open Cycle Cooling Water Program B.1.13 and Aging Management Reference 3.3.2.300, Bolting Integrity Program B.1.12. The refuse pit is addressed in LRA Table 2.4-11 under component groups "Concrete Walls and Concrete Slabs" and Aging Management Reference 3.5.1.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31.

- Discharge Outfall Structure – This structure is addressed in LRA Table 2.3.3-22, under Component Groups, "Concrete Walls and Concrete Slabs", and Aging Management Reference 3.5.1.22, RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Program B.1.31 (as supplied in RAI 2.4-7 response). However, these aging management activities are an enhancement and are not currently implemented.

- 8' Diameter Ice Melt Recirculating Pipe, including ice melt gate – The ice melt pipe is addressed in LRA Table 2.3.3-22, under Component Group, "Piping and Fittings", and Aging Management Reference 3.3.1.15, Open Cycle Cooling Water Program B.1.13. The gate is addressed in LRA Table 2.3.3-22, under Component Group, "Valves", and Aging Management Reference 3.3.2.278, Open Cycle Cooling Water Program B.1.13. and Aging Management Reference 3.3.2.300, Bolting Integrity Program B.1.12.

- Discharge flume/canal – The aging management for this earthen structure is discussed in the response to (1) above.

(3) Section B.1.31 of the LRA does list the operating experience for concrete structures. Section B.1.13 of the LRA lists the operating experience for the piping components covered by the Open Cycle Cooling Water Program.

The operating experience for the earthen structures:

Dresden has performed inspections of the intake and discharge canals and has not found any appreciable silting. However, minor silting was found at the intake structure near the bar racks. This silting was removed prior to loss of function of the ultimate heat sink.

Quad Cities, taking suction directly off the Mississippi River, has found significant levels of silting in the earthen structure of the intake flume as well as at the intake structure on several occasions. Timely corrective actions (dredging or cleaning) were completed prior to the loss of function of the ultimate heat sink, indicating an effective monitoring program.

(4) The only enhancement needed is to annotate the existing requirements of the Exelon procedures that monitor the aging effect, "Loss of Form due to Sedimentation", to ensure that the required volume of water is available in the ultimate heat sink to support emergency cooling conditions as license renewal commitments. This requirement is implemented through a site Predefine Activity for scheduling and tracking purposes at Quad Cities. A similar Predefine Activity will be developed for Dresden.

(5) The Quad Cities intake flume is in the scope of license renewal. Management of this earthen structure is discussed in the response to (1) above.

(6) A baseline inspection will be performed prior to the period of extended operation for the Quad Cities Discharge Flume/Canal and Weir Gate and the Dresden Outfall Structure. Any problems affecting performance of intended functions will be identified and corrected.

Note that there are no current aging management activities performed for the Dresden Stop Logs. As there has been no viable aging mechanism identified, the stop logs will not be included in the baseline inspection.

(7) LRA Section A.1.31 for Dresden and Quad Cities will be updated to specifically indicate the applicable in-scope earthen structures (Dresden intake/discharge flumes and Quad Cities intake/discharge flumes) and the aging management activities associated with these structures.