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Nuclear

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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.

(3) 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 21, 2003.

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(4) 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 December 5, 2003.

(5) 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 December 12, 2003.

Exelon Generation Company, LLC (EGC) is submitting the additional information requested in email requests sent by Tae Kim (NRC) to EGC on October 14 and 23, 2003, and November 5 and 25, 2003 and in a teleconference on December 8, 2003. This additional information provides a response to questions regarding Sections 2.1, 2.3, 3.1, 3.3, 3.6, and the Aging Management Programs sections of Reference 1. In addition, EGC is revising the responses to Request for Additional Information (RAI) B.1.2 that was submitted in Reference 2, RAI 3.1-11 that was submitted in Reference 3, RAI 2.3.4.2-3 that was submitted in Reference 4, and RAI B.2.2-1 that was submitted in Reference 5.

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

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

Respectfully,

12/22/03
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing

Attachment : Response to Request for Additional Information – LRA Sections 2.1, 2.3, 3.1, 3.3, 3.6, and 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 2.1, 2.3, 3.1, 3.3, 3.6, and Aging Management Programs

RAI 2.1-2 Supplemental Information Request

In discussions with the applicant's license renewal project team, the NRC staff noted some cases where nonsafety-related plant equipment was credited with providing anchorage for nonsafety-related piping that was attached to safety-related piping. In these cases, the nonsafety-related piping was placed within the scope of license renewal, but the plant equipment providing structural support was not considered to be within scope. For cases where an entire pipe run including both safety and nonsafety-related piping was analyzed as part of the current licensing basis to establish that it could withstand design basis event loads, NUREG-1800, Section 2.1.3.1.2 indicates that the scoping methodology includes: (1) the nonsafety-related piping up to its anchors, and (2) the associated piping anchors as being within the scope of license renewal under 10 CFR 54.4(a)(2). Because the plant equipment credited with providing support to nonsafety-related piping within the scope of license renewal appears to be equivalent to an associated piping anchor as described in NUREG-1800, the staff requested the applicant to provide justification for not including this plant equipment within the scope of license renewal.

In their October 3, 2003 response to RAI 2.1-2, the applicant stated that they conservatively included those portions of non-safety related pipe up to the point where the pipe was restrained in three orthogonal directions. The scoping boundary was determined through a review of isometric pipe drawings. In those instances where isometric drawings of non-safety related pipe did not exist (typically small bore pipe less than 2 ½ inches in diameter), Exelon either included the entire line up to the end of the pipe run (e.g., no more pipe existed) or ended the boundary where the line attached to a larger piping header or a major component (i.e., pump or heat exchanger). The larger piping header or major component was treated as an anchor. However, the applicant stated that the major component was excluded from the scope of License Renewal because all pipe supports installed in the plant were included within the scope of License Renewal.

The staff determined that the applicant did not provide a sufficient basis for excluding major components credited with providing a pipe support function from the scope of license renewal. The staff concluded that major components that ensure satisfactory accomplishment of a safety-related function by providing support to nonsafety-related piping attached to safety-related systems should be included within the scope of license renewal. The staff noted that the intended function performed by these major components is similar to that performed by pipe supports. This issue is identified as Open Item 2.1-2.

Response

In those instances where isometric drawings of non-safety related pipe attached to safety related pipe did not exist (typically small bore pipe less than 2 ½ inches in diameter), Exelon either included the entire line up to the end of the pipe run (e.g., no more pipe existed) or ended the boundary where the line attached to a larger piping header or a major component (i.e., pump or heat exchanger). The original scoping results submitted in the License Renewal Application did not include these components within the scope of license renewal. As a result of further review, Exelon has decided to add these components into the scope of license renewal as non-structural components that provide non-safety related anchorage. Exelon subsequently performed a review of the boundary diagrams for each site to identify those major components and larger piping headers that were credited as an anchor for non-safety related piping. The results of this review along with the applicable aging management are contained in Tables 1 and2 below.

To encompass these components the following changes will be made to LRA Table 2.4-15 (Component Groups Requiring Aging Management Review – Component Supports) and to LRA Table 3.5-2 (Aging management review results for containments, structures and component supports that are not addressed in NUREG-1801).

- A new Component group will be added to LRA Table 2.4-15. The new Component category will be "Non-Structural Components Providing Non-SR Anchorage." The Component Intended Function for this Component will be "Non-SR Structural Support." The components that roll-up to the new Component category will be those components listed in Tables 1 and 2 below.

- A new Aging Management Reference will be added to Table 3.5-2 that is applicable to this component group. The Table 3.5-2 line item for this Aging Management Reference will be –

Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
Non-Structural Components Providing Non-SR Anchorage	Various	Various	Loss of material/General corrosion	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address Non-Structural Components Providing Anchorage for Non-Safety Related Piping.

Table 1
Dresden Non-Structural Components Credited Solely as an Anchor for Non-Safety Related Pipe Attached to Safety Related Pipe

Component name	Component ID #	Boundary Diagram #	Component Intended Function	LRA Aging Management Reference(s)
Unit 2 & 3 high pressure feedwater heaters	2(3)-3105-D1, D2, D3	LR-DRE-M-14 LR-DRE-M-347	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 & 3 reactor water cleanup auxiliary pump	2(3)-1206	LR-DRE-M-30 LR-DRE-M-361	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 & 3 standby liquid control test tank	2(3)-1104	LR-DRE-M-33 LR-DRE-M-364	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 & 3 traversing in-core probe (TIP) chamber shield	2(3)-0737-A/B/C/D/E (Not shown on boundary diagrams)	LR-DRE-M-37-2 LR-DRE-M-367-2	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 reactor building service air supply line	2-4609-4"-O	LR-DRE-M-38-2	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 & 3 reactor building vent & drain header lines	2-4891-8"-LX 2-0222A-8"-LX 2-4808-6"-LX 2-4812-8"-LX 2-4827-8"-LX 3-4827-4"-LX 3-4808-6"-LX 3-0222A-8"-LX	LR-DRE-M-39 LR-DRE-M-369	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2/3 control room ventilation exhaust fans	2/3-57548 2/3-57545	LR-DRE-M-273-2	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 2 & 3 Condensate header lines	2(3)-3316-20"-L	LR-DRE-M-366 LR-DRE-35-1	Non-SR Structural Support	See change in LRA Table 3.5-2, above.

Table 2
Quad Cities Non-Structural Components Credited Solely as an Anchor for Non-Safety Related Pipe Attached to Safety Related Pipe

Component name	Component ID #	Boundary Diagram #	Component Intended Function	LRA Aging Management Reference
Unit 1 & 2 high pressure feedwater heaters	1(2)-3105-D1, D2, D3	LR-QDC-M-15-1 LR-QDC-M-62-1	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 1 & 2 condensate piping	2-33445-4"-L 1-33128-4"-L	LR-QDC-M-16-5	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 1 & 2 standby liquid control test tank	1(2)-1104	LR-QDC-M-40 LR-QDC-M-82	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 1 & 2 reactor building vent and drain header piping	1(2)-4891-8"-L 1(2)-4827-8"-L 1(2)-4811-8"-L 1(2)-0220-8"-L 1(2)-2029-12"-L	LR-QDC-M-43 LR-QDC-M-85	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 1 & 2 drywell pneumatic air compressors	1(2)-4708	LR-QDC-M-24-12 LR-QDC-M-71-7	Non-SR Structural Support	See change in LRA Table 3.5-2, above.
Unit 1 & 2 Traversing In-core probe (TIP) chamber shields	1(2)-0734B-F	LR-QDC-M-584-1 LR-QDC-M-584-2	Non-SR Structural Support	See change in LRA Table 3.5-2, above.

RAI 2.3.1.2-5 Supplemental Information Request

In response to RAI 2.3.1.2-5, the applicant identifies the following components that are included in jet pump assemblies: thermal sleeve, inlet header, riser brace arm, hold down beams, inlet elbow, mixing assemblies, and diffuser. The staff compared the applicant response with the list of BWR jet pump assembly components that are within the scope of license renewal as identified in Appendix A (Section A.2) of BWRVIP-41. The staff finds that the applicant has not identified the following four components: transition piece, riser pipe, adapter, and restrainer bracket. Provide an explanation for not including these four components in Component Group, "Jet Pump assemblies."

Response:

The transition piece, riser pipe, adapter, and restrainer bracket are included in the Component Group "Jet Pump Assemblies." The previous response considered the transition piece and riser pipe to be part of the inlet header, and the adapter as part of the diffuser. The restrainer bracket was not specifically identified, but is part of the assembly.

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 are 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, ASME Section XI Aging Management Program B.1.1 for the remaining portion (safe-end, cap, and welds), and Water Chemistry as stated in Aging Management Program B.1.2.

- 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 ASME Section XI, category B-F, and the safe end-to-cap weld is ASME Section XI, category B-J. These welds are GL 88-01, category A welds. All GL 88-01, category A welds were subsumed into the Risk Informed ISI Program as noted on the Relief Request Approved by the NRC on ADAMS Accession Number ML012050103. Similar Relief Requests have been submitted for the next 10 Year Inspection Periods at Dresden and Quad Cities. Therefore, none of the welds listed below are in the scope of BWRVIP-75 or GL 88-01.

- 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 3.1-7 Supplemental Information Request

(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. SER for BWRVIP-26 states that the threshold fluence level for IASCC is 5 x 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 x 10^{20} n/cm^2 (E > 1 MeV)] by the end of extended period of operation are susceptible to cracking due to IASCC. Per SER for BWRVIP-26, the accumulated neutron fluence is a TLAA issue for these vessel internals. The SER for BWRVIP-26 further states that the applicant must identify and evaluate this TLAA issue. Provide identification and evaluation of the accumulated neutron fluence received by the D/QNPS vessel internals at the end of the extended license period as a TLAA issue.

Response:

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

General Electric Nuclear Energy Service Information Letter SIL 409, "Incore Dry Tube Cracks," identified the occurrence of cracking of incore dry tubes / guide tubes due to a combination of crevice induced IGSCC with crack propagation by irradiation assisted stress corrosion cracking (IASCC). The guidance in SIL 409 is utilized by Dresden and Quad Cities stations to schedule and perform inspections of the incore dry tubes. SIL 409 recommends that the upper two feet of a dry tube be inspected during the refueling outage following the twentieth calendar year after dry tube replacement, with subsequent inspections every fourth calendar year following the initial inspection. SIL 409 recommends that dry tubes with detected cracks be replaced. Inspection of SRM & IRM dry tubes / guide tubes to detect the aging effects of IASCC are included in Exelon aging management program B.1.9, BWR Vessel Internals. Inspection of dry tubes that have not been replaced are performed in accordance with the recommendations of SIL 409 during each refueling outage. Exelon has replaced incore dry tubes during refueling outages. When a dry tube is replaced, the inspection interval for the replacement dry tube is extended to 20 years. After the 20 year inspection has been completed, additional inspections are performed once every 4 years.

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

RAI 3.1-11 Supplemental Information Request

The applicant's response to RAI 3.1-11 states that the Aging Management program B.2.6 manages loss of material and crack initiation and growth in the Dresden isolation condensers. This program applies to the tubing, tubesheet, channel heads and shells, and consists of performing eddy current testing of the tubes as well as temperature and radiation monitoring of the shell-side water (which are consistent with NUREG-1801). However, NUREG-1801, items IV.C.1.4-a and b requires the following two enhancements in addition to the AMPs in Chapter XI.M1 and XI.M2: (1) augmented inspections to detect cracking due to SCC and cyclic loading or loss of material due to pitting and crevice corrosion in isolation condenser components (tubing, tubesheet, channel head, and shell, and (2) verification of the effectiveness of the program. NUREG-1801 identifies temperature and radioactivity monitoring of the shell-side water, and eddy current testing of the tubes as a verification program. This verification program does not verify that no cracking or loss of material is occurring in the isolation condenser shell, channel head and tubesheet. Therefore, provide augmented inspection of the Dresden isolation condenser (i.e. VT or UT) to manage loss of material, and crack initiation and growth in the isolation condenser tubesheet, channel head, and shell, as required by NUREG-1801.

Response

The response is divided into two parts. For clarity of understanding, background information on the RAI is first provided. Afterwards, the response to the supplemental information request is provided.

Background information for RAI 3.1-11 Supplemental Information Request

Aging analysis of the isolation condenser can be found in Section IV C1 in NUREG 1801. This section was intended for systems connected to the pressure vessel extending outward to the second containment isolation valves. The Dresden isolation condensers are installed outboard of the second containment isolation valve. As a result, the isolation condenser are not classified as ISI Class 1, but are classified as ISI Class 2 on the tube side, and ISI Class 3 on the shell side. As described in the License Renewal Application (LRA) Section 2.3.2.5, the isolation condenser is a heat exchanger which consists of two tube bundles immersed in a large storage tank, with the tank vented to the atmosphere.

NUREG 1801, items IV.C.1.4-a and b, require that the aging management program (AMP) in Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" for Class 1 components be augmented to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion. It also requires that the effectiveness of AMP XI.M1 be verified. It states that an acceptable verification program is to include temperature and radiation monitoring of the shell side water, and eddy current testing of the tubes.

LRA Table 3.1-1, items 3.1.1.2 (loss of material due to general, pitting, and crevice corrosion) and 3.1.1.7 (crack initiation and growth due to stress corrosion cracking or cyclic loading) credited the inservice inspection and water chemistry AMPs as described in the LRA Appendix B, Sections B.1.1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and B.1.2, "Water Chemistry." Items 3.1.1.2 and 3.1.1.7 included discussions of further evaluations in LRA Sections 3.1.1.1.2 and 3.1.1.1.7 respectively. Both Sections 3.1.1.1.2 and 3.1.1.1.7 stated that AMP B.1.1 will be augmented by AMP B.2.6, and that B.2.6 activities include temperature and radiation monitoring of the shell side water, and eddy current testing of the tubes, to ensure intended function is maintained.

The initial RAI 3.1-11 stated that "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."

The response to RAI 3.1-11 stated that "Aging management program B.2.6, "Heat Exchanger Test and Inspection Program," manages loss of material and crack initiation and growth in the Dresden isolation condensers. This program, as it applies to the isolation condensers, which include the tubing, tube sheets, channel heads and shells, consists of performing eddy current testing of the tubes as well as temperature and radiation monitoring of the shell-side (cooling) water."

This supplemental information request to RAI 3.1-11 stated that "NUREG-1801 identifies temperature and radioactivity monitoring of the shell-side water, and eddy current testing of the tubes as a verification program. This verification program does not verify that no cracking or loss of material is occurring in the isolation condenser shell, channel head and tubesheet. Therefore, provide augmented inspection of the Dresden isolation condenser (i.e. VT or UT) to manage loss of material, and crack initiation and growth in the isolation condenser tubesheet, channel head, and shell, as required by NUREG-1801."

RAI 3.1-11 Supplemental Information Request Response

Aging management program (AMP) B.2.6, "Heat Exchanger Test and Inspection Program," is a ten-element program that was developed to address heat exchangers in scope of license renewal that are not inspected under other AMPs. The intent of the AMP B.2.6, as originally developed and described In the LRA, is to require a visual inspect of the isolation condenser channel head, tube sheet, and shell, in addition to performing eddy current testing of the tubes, and temperature and radiation monitoring of the shell-side water. These are all new activities that will be implemented prior to the period of extended operation.

License Renewal Application, Appendix B, Section B.2.6, summarized Aging management program B.2.6, "Heat Exchanger Test and Inspection Program," but did not clearly describe the visual inspection of the isolation condenser tubesheet, channel head, and shell in the description of the isolation condenser augmented activities.

In addition to identifying the augmented isolation condenser inspection activities of temperature and radiation monitoring of the shell-side water, and eddy current testing of the tubes, AMP B.2.6 provides for condition monitoring, inspection, and performance testing of heat exchangers in scope of license renewal that are not inspected under other AMPs, including the isolation condensers.

Aging management program AMP B.2.6 requires that the isolation condenser be inspected in according with the station heat exchanger inspection program, and eddy current tested in accordance with corporate procedures. These inspections and testing are performed by qualified inspectors.

(1) In conjunction with the periodic eddy current testing of the tubes, a visual inspection to detect cracking and loss of material of the channel head and tube sheets will be performed on the tube-side of the isolation condenser in accordance with the station heat exchanger inspection program as an augmented inspection to manage loss of material and crack initiation and growth in the isolation condenser tube sheet and channel head.

(2) Shell-side visual inspections are presently periodically performed to verify the integrity of shell-side internal structural components. These inspections will be expanded in accordance with the station heat exchanger inspection program to visually inspect the shell to detect cracking and loss of material of the shell as an augmented inspection to manage loss of material and crack initiation and growth in the isolation condenser shell.

LRA Appendix A, Dresden Units 2 and 3, Section A.2.6, "Heat Exchanger Test and Inspection Activities," paragraph 3, is revised as follows to include the discussion of the visual inspection of the isolation condenser tube sheet, channel head, and shell.

"The isolation condenser test and inspection augmentation activities detect cracking due to stress corrosion cracking or cyclic loading, and detect loss of material due to pitting and crevice corrosion. These are ISI augmentation activities, outside the ISI program, not augmented ISI activities within the ISI program. These augmentation activities verify that significant degradation is not occurring, and therefore that the intended function of the isolation condenser is maintained during the extended period of operation. These augmentation activities consist of temperature and radioactivity monitoring of the shell-side (cooling) water, eddy current testing of the tubes, and visual inspections of the channel head, tube sheets, and internal surfaces of the shell."

RAI 3.1-21a Supplemental Information Request

The applicant's response to RAI 3.1-21a states that properly controlled water chemistry is adequate to manage cracking due to SCC and inspection for occurrence of SCC in the CRD system is not required. This response is not acceptable because the applicant has not provided a program for verifying the effectiveness of the Water Chemistry Program for providing adequate protection against SCC. Provide an aging management program to verify that the Water Chemistry Program is providing adequate protection against cracking due to SCC.

Response

The process fluid temperature in the control rod drive (CRD) hydraulic system is less than 100 deg-F, and the typical flow conditions are either low flow (in the cooling water line) or stagnant flow (in the charging water and drive water lines). With process temperatures below 140 deg-F, EPRI TR 1003056 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 CRD hydraulic system 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 CRD hydraulic system.

3.1-24d Supplemental Information Request

(i) In its response to RAI 3.1-24d, the applicant commits to inspection intervals for Category C through E welds at Quad Cities, Units 1 and 2, that are consistent with the requirements of BWRVIP-75 for normal water chemistry. However, NRC SER of EPRI Report TR-113932 (BWRVIP-75), dated May 14, 2002 expanded on the guidelines and inspection frequencies for Category C welds to include plants that comply with BWRVIP-61 and those plants that do not comply with BWRVIP-61. Therefore, confirm whether or not the D/QCNPS plants are complying with BWRVIP-61 and that the appropriate inspection frequencies based on the NRC SER are used. Also, identify the number of welds in each category of weld that are credited for the use of IHSI, HWC, NMCA or a combination of these methods, and the corresponding inspection frequency. Provide the number of Category C through E welds and the frequency of their inspections for Quad Cities 1 and 2.

(ii)In response to RAI 3.1-24d, the applicant states that HWC/NMCA inspection frequencies for Categories C through E welds were reduced for Dresden, Unit 2 and only applied to those weld locations where the improved water chemistry is effective. Explain how these locations were identified. Explain the two different categories for C, D, and E, and why two different inspection frequencies are listed in the response for each Category C, D, and E welds at Dresden, Unit 2. In addition, confirm whether the information provided meet the requirements of BWRVIP-75, as approved by NRC SER of EPRI Report TR-113932 (BWRVIP-75), dated May 14, 2002 (i.e. RAI response states that Category D-HWC welds with a population of 24 received 10% inspection every 6 years, while BWRVIP-75 requires 100% every 10 years for HWC).

(iii)In response to RAI 3.1-24d, the applicant further states that Category A welds at D/QCNPS are inspected per the RISI guidelines. Confirm that Category A welds at D/QCNPS are inspected to BWRVIP-75 as modified and approved by NRC SER of EPRI Report TR-113932 (BWRVIP-75), dated May 14, 2002.

(Iv) BWRVIP-75 states that the HWC program must be effective to qualify for the reduced inspections and the methodology used is provided in BWRVIP-62. In response to RAI 3.1-24c, the applicant used the term "HWC index" to determine the factor of improvement on crack growth. Clarify how the "HWC index" is used to determine factor of improvement? Does this term, "HWC index", mean the availability of HWC at a certain ECP value?

Response

(i) Part (i) of this RAI Supplemental Information Request contains 3 separate requests. These are:

- Confirm whether or not the D/QCNPS plants are complying with BWRVIP-61 and that the appropriate inspection frequencies based on the NRC SER are used.
- Identify the number of welds in each category of weld that are credited for the use of IHSI, HWC, NMCA or a combination of these methods, and the corresponding inspection frequency.
- Provide the number of Category C through E welds and the frequency of their inspections for Quad Cities 1 and 2.

The response to each of these questions is provided below.

Request 1:
Confirm whether or not the D/QCNPS plants are complying with BWRVIP-61 and that the appropriate inspection frequencies based on the NRC SER are used.

Response 1:
Dresden and Quad Cities comply with the conditions of BWRVIP-75 that permit reductions in the frequencies for inspection of Category C welds (non-resistant materials, stress improved after 2 years of operation) from 100 percent every 10 years to 25 percent every ten years under NWC (normal water chemistry) and 10 percent every 10 years when HWC (hydrogen water chemistry) is implemented.

The NRC Final Safety Evaluation of BWRVIP-75 (TAC No. MA5012) dated May 14, 2002 imposed changes to BWRVIP-75 for Category C welds. Specifically, in order to utilize the reduced frequencies specified in BWRVIP-75, the following had to be performed:

"The Owner must ensure that an effective stress improvement was achieved performed [sic]. Additionally, there must have been either:

a) a preservice (post-stress improvement) and inservice examination with a qualified procedure with no cracking identified, or
b) for welds that were previously stress-improved but did not receive a preservice examination, at least one examination performed with a qualified procedure after more than within [sic] two operating cycles of the licensee's adoption of this guidance and no cracking detected."

The SER imposed further restrictions on improvement in inspection frequencies for plants with Category C welds that had been treated with Induction Heating Stress Improvement (IHSI) but did not fully comply with BWRVIP-61. These further restrictions (specified in the initial SER for BWRVIP-75 dated September 15, 2000) only apply to plants that credited the IHSI process for stress improvement (IHSI is the specific subject of BWRVIP-61). The staff explained in the September 2000 SER that these further restrictions applied to "**plants that used IHSI** to mitigate IGSCC, but do not fully comply with the recommendations of the BWRVIP-75 report."

The Category C welds at Dresden and Quad Cities were stress improved by the Mechanical Stress Improvement method (MSIP). This process was accepted by the SER on BWRVIP-75 without restrictions providing that effective stress improvement and confirmatory inspections were performed as stated above. Dresden and Quad Cities are in compliance with the BWRVIP-75 requirements. Therefore the reduced inspection frequencies of Category C welds apply to each site. The appropriate inspection frequencies are being used.

Since neither Dresden or Quad Cities has any IHSI-improved Category C welds, the requirements of BWRVIP-61 do not apply to these plants. Therefore the only required compliance is to BWRVIP-75 as described above.

<u>Request 2:</u>
Identify the number of welds in each category of weld that are credited for the use of IHSI, HWC, NMCA or a combination of these methods, and the corresponding inspection frequency.

<u>Response 2:</u>
There are no welds credited for use of IHSI, HWC, NMCA, or any combination of these methods at Quad Cities or Dresden Unit 3. As indicated above, there are no welds credited for IHSI at Dresden Unit 2 since MSIP was used for stress improvement. The number of welds in each category at Dresden Unit 2 that are

credited for use of HWC/NMCA, and the corresponding inspection frequencies are as identified below:

Category	Total Population	Inspection Frequency
Unit-2:		
C	28	10% every 10 years
D	40	100% every 10 years, 50% in first six years
E	37	10% every 10 years

Request 3:
Provide the number of Category C through E welds and the frequency of their inspections for Quad Cities 1 and 2.

Response 3:
The number of Quad Cities Units 1 and 2 Category C through E welds and associated inspection frequencies are as identified below:

Category	Total Population	Inspection Frequency
Unit-1:		
C	123	25% every 10 years, at least 50% in first 6 years
D	4	100% every 6 years
E (Overlay)	42	25% every 10 years, at least 50% in first 6 years
E (MSIP*)	2	100% every 6 years
Unit-2:		
C	129	25% every 10 years, at least 50% in first 6 years
D	0	NA
E (Overlay)	38	25% every 10 years, at least 50% in first 6 years
E (MSIP*)	0	NA

*MSIP = Mechanical Stress Improvement

(ii) The locations for which Dresden Unit 2 Category C through E weld inspection frequencies were reduced are those areas in the reactor coolant flowpath. These portions of piping are continually exposed to circulating reactor coolant and receive the benefits of IGSCC mitigation due to HWC/NMCA.

There were a number of typographical errors in the Unit 2 weld information provided in the original response to RAI 3.1-24d. The same errors were provided in the response to related RAI B.1.7a. These errors resulted in a perception that different frequencies were provided for each weld category and/or that inappropriate frequencies were used. The Unit 2 portion of the associated tables in the original response to RAI 3.1-24d should have read as follows (corrections provided in bold and underlined):

Category	Total Population	Welds Inspected
Unit-2:		
C-HWC	28	3 (10% every 10 years)
C-NWC	66	17 (25% every 10 years)
D-HWC	40	40 (100% every 10 years, 50% in first six years)
D-NWC	24	24 (100% every 6 years)
E-HWC	37	4 (10% every 10 years)
E-NWC	1	1 (25% every 10 years)

The inspection frequencies for the Dresden Unit 2 welds as identified above meet the requirements identified in the NRC SER of EPRI Report TR-113932 (BWRVIP-75).

(iii) IGSCC Category A welds are subsumed under the EPRI Risk-Informed Inservice Inspection (RI-ISI) program. This is consistent with the methodology of EPRI Report TR-112657, Revision B-A, Revised Risk-Informed Inservice Inspection Evaluation Procedure.

(iv) The term "HWC index" used in the response to RAI 3.1-24c is the availability (in percent) of HWC System at a certain ECP value. The availability is calculated as the percentage of time the feedwater hydrogen concentration is sufficient to achieve an ECP value of ≤-230 mV (standard hydrogen electrode) and the reactor water temperature exceeds 200°F. Factor of improvement (FOI) is then determined per Section 5.5, Factor of Improvement (Calculational Basis) of EPRI TR-103515-Rev. 2, BWR Water Chemistry Guidelines, 2000 Revision.

Dresden and Quad Cities do not use the FOI approach identified in BWRVIP-75 to determine the effectiveness of HWC.

RAI 3.3.2.4.24(a) Supplemental Information Request

Loss of material from selective leaching may be an applicable aging effect/aging mechanism for cast iron and brass components in saturated steam/condensate as well as air, moisture, humidity, and leaking fluid environments if stagnant liquids are present in these environments; however, the LRA only identifies a loss of material due to general corrosion for these components. By letter dated August 4, 2003, the staff requested, in RAI 3.3.2.4.24(a), the applicant to clarify whether selective leaching is applicable to components in the plant heating system and, if so, to provide the applicable AMP(s). In its response dated October 3, 2003, the applicant stated that in these components are subject to loss of material due to general corrosion, and they may also be subject to loss of material due to sélective leaching if stagnant liquids are present in these environments. The loss of material is managed by a one-time inspection, which requires the inspection of a representative sample of components in these environments to detect for signs of degradation. The applicant further stated that selective leaching in brass alloys results in either a uniform attack or a localized plug attack, while selective leaching of gray cast iron results in iron being dissolved, leaving a porous mass consisting of graphite, voids, and rust. The applicant stated that the one-time inspection will detect the loss of material whether it is due to selective leaching or general corrosion. The response to RAI 3.3.2.4.24(a) states that the OTI will be conducted on a representative sample of components in "these environments." Clarify whether "these environments" include locations where liquids would pool, such that potential selective leaching would be identified.

Response

In the response to RAI B.1.23-2, Exelon has agreed to perform periodic inspections of plant heating steam components rather than perform one-time inspections. These inspections will be performed at a frequency not to exceed once every five years. The philosophy for one-time inspections was to select components in stagnant flow areas where practical since those locations are most likely to experience loss of material due to the identified aging mechanisms. The same approach will apply to the periodic inspections of the heating steam system discussed in the response to RAI 3.3.2.4.24(a). A new UFSAR supplement section for each station for this AMP is included as an attachment to the response to RAI B.1.23.

RAI 3.6-09 Supplemental Information Request

In the response to RAI 3.6-09, the licensee stated that they would develop a program that is consistent with NUREG 1801 aging management program XI.E.2 to manage electrical cables not subject to 10 CFR 50.49 environmental qualification requirements used in nuclear instrumentation circuits. The staff requests that the licensee provide a description of this program in comparison to NUREG 1801 along with a copy of the UFSAR supplement for this new program.

Response

Exelon will implement aging management program B.1.37, Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits.
The purpose of this aging management program is to provide reasonable assurance that the intended function of cables used in instrumentation circuits with sensitive, low-level signals which are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation. The cables included within the scope of this program are the cables used in the following Nuclear Instrumentation systems (NIS) and radiation monitoring systems:

- Source Range Monitors (SRM)
- Intermediate Range Monitors (IRM)
- Local Power Range Monitors (LPRM)
- Drywell High Range Radiation Monitors
- Main Steam Line (MSL) Radiation Monitors
- Steam Jet Air Ejector (SJAE) Radiation Monitors

The following is the LRA Appendix A and B revision for the aging management program electrical cables not subject to 10CFR 50.49 environmental qualification requirements used in instrument circuits.

A.1.37 Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits - Dresden

The cables of the Nuclear Instrumentation systems which includes Source Range monitors (SRM), Intermediate Range Monitors (IRM), Local Power Range Monitors (LPRM), and Radiation Monitoring systems which includes Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors are sensitive instrumentation circuits with low-level signals and are located in areas where the cables could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Calibration testing, cable testing or surveillance tests is performed to ensure that the cable insulation resistance is adequate for the instrumentation circuits to perform their intended functions. This provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance and cable testing. This aging management program is a new program. The calibration testing, cable testing and surveillance testing that will be used for this program are performed currently, and are effective in identifying the existence of age related degradation. The program will be implemented prior to the period of extended operation and will include a review of the calibration and surveillance results for cable aging degradation.

A.1.37 Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits – Quad Cities

The cables of the Nuclear Instrumentation systems which includes Source Range monitors (SRM), Intermediate Range Monitors (IRM), Local Power Range Monitors (LPRM), and Radiation Monitoring systems which includes Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors are sensitive instrumentation circuits with low-level signals and are located in areas where the cables could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Calibration testing, cable testing or surveillance tests is performed to ensure that the cable insulation resistance is adequate for the instrumentation circuits to perform their intended functions. This provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance and cable testing. This aging management program is a new program. The calibration testing, cable testing and surveillance testing that will be used for this program are performed currently, and are effective in identifying the existence of age related degradation. The program will be implemented prior to the period of extended operation and will include a review of the calibration and surveillance results for cable aging degradation.

B.1.37 Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Description

The aging management program for electrical cables not subject to 10CFR 50.49 environmental qualification requirements used in instrument circuits within the scope of License Renewal provides aging management for cables used in sensitive instrumentation circuits with low-level signals.

The cables of the Nuclear Instrumentation systems, which include Source Range monitors (SRM), Intermediate Range Monitors (IRM), Local Power Range Monitors (LPRM); and Radiation Monitoring systems, which include Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors are sensitive instrumentation circuits with low-level signals and are located in areas where the cables could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. This program considers the technical information and guidance provided in NUREG/CR-5643, Insights Gained From Aging Research, U.S. Nuclear Regulatory Commission, March 1992; IEEE Std. P1205-2000, IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations; SAND96-0344, Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, September 1996; and EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments, Electric Power Research Institute, Palo Alto, CA, June 1999.

The program is implemented through station procedures that are used to perform calibration testing on the LPRM, Drywell High Range Radiation Monitors, Main Steam Line, Radiation Monitors, and Steam Jet Air Ejector Radiation Monitors systems. When an instrumentation channel is found to be out of tolerance or out of calibration, corrective action is taken such as recalibration and circuit trouble-shooting of the instrumentation cable system. A review of the calibration results for cable aging degradation will be performed before the period of extended operation and every 10 years thereafter.

Station procedures are used to perform surveillance testing and cable testing that is effective in determining cable insulation condition on the SRM and IRM systems. Corrective action, such as cable replacement, will be taken if a cable fails to meet the acceptance criteria of the cable or surveillance test. A review of the surveillance results for cable aging degradation will be performed before the period of extended operation and every 10 years thereafter.

NUREG-1801 Consistency

The "Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits" aging management program is consistent with the ten elements of aging management program XI.E2, "Electrical Cables Not Subject To 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits " specified in NUREG-1801 with the following exceptions. The program will be implemented prior to the period of extended operation.

Exceptions to NUREG-1801

NUREG-1801 requires calibration tests be performed as part of the plant technical specifications and are specific to the instrumentation loop being calibrated. For the Source Range Monitors (SRM) and Intermediate Range Monitors (IRM) systems, cable and surveillance testing is used to detect aging of the cable prior to loss of intended function. Cable and surveillance testing is acceptable in detecting aging degradation prior to the loss of cable intended function and is consistent with the proposed interim staff guidance (ISG)-15.

Enhancements

- ISG-15 requires a review of the calibration results for cable aging degradation once every 10 years. This program will be revised to include a review of the LRPM, Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors calibration results for cable aging degradation before the period of extended operation and every 10 years thereafter.

- ISG-15 requires a review of the surveillance results for cable aging degradation once every 10 years. This program will be revised to include a review of the IRM and SRM surveillance results for cable aging degradation before the period of extended operation and every 10 years thereafter.

Operating Experience

This program is new. Therefore, no programmatic operating experience is available. The surveillance testing and calibration that will be used for this program are performed currently, and are effective in identifying the existence of age related degradation.

Conclusion

This aging management program "Electrical Cables Not Subject to 10CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits" provides reasonable assurance that aging effects are adequately managed so that the intended functions of these types of cables are maintained during the period of extended operation.

Program Comparison Against NUREG 1801

Attached below is the full 10 element comparison of the requirements contained in NUREG 1801 aging program XI-E-2 against Exelon aging management program B.1.37.

Program Elements	XI.E2, Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits NUREG-1801 Description	(B.1.37) Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Dresden-Quad Cities Basis	GALL Exception	Comments
Program Description	In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environment may be more severe than the plant design environment for those areas. Conductor insulation material used in electrical cables may degrade more rapidly than expected in the adverse localized environment. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have immediate adverse effect on operability. Exposure of electrical cables to adverse localized environments caused by heat or radiation can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument loop. The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10 CFRR 50.49 and are used circuits with sensitive, low-level signals exposed to adverse localized environments cause by heat, radiation, or moisture will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619. In this aging management program, routine calibration tests performed as part of the plant surveillance test program are used to identify the potential existence of aging degradation. When an instrumentation loop is found to be out of calibration during routine surveillance testing, troubleshooting is performed on the loop, including the instrumentation cable. As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is the failures of deteriorated cable systems (cables connections, and penetrations) might be induced during accident conditions. " Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the electrical cables covered by the aging management program are either not exposed to harsh accident conditions or are not required to remain function during or following an accident to which they are exposed.	For Dresden and Quad Cities Stations, the cables within the scope of this program are the cables used in sensitive instrumentation circuits with low level signals of the Nuclear Instrumentation systems (NIS) which includes Source Range Monitors (SRM), Intermediate Range Monitors (IRM), Local Power Range Monitors (LPRM), and Radiation Monitoring systems which includes Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors. The purpose of this aging management program is to provide reasonable assurance that the intended function of cables used in instrumentation circuits with sensitive, low-level signals, which are not subject to the environmental qualification requirements of 10 CFRR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation. In most areas, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environment may be more severe than the plant design environment for those areas. For Dresden and Quad Cities, adverse conditions are expected to be present inside the drywell when compared to the reactor and turbine buildings. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619. In this aging management program, calibration testing, cable testing or surveillance tests will be credited to ensure that the cable insulation resistance is adequate for the instrumentation circuits to perform their intended functions. When an instrumentation channel is found to be out of calibration during routine surveillance testing, troubleshooting is performed on the loop, including the instrumentation cable. For Dresden and Quad Cities the cable systems covered by this aging management program are not EQ and are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.	Yes	1, 2, 3

	XI.E2 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits	(B.1.37) Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits		
Program Elements	NUREG-1801 Description	Dresden-Quad Cities Basis	GALL Exception	Comments
1.Scope of Program	*This program applies to electrical cables used in circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation that are within the scope of license renewal.*	The Dresden and Quad Cities instrumentation cables within the scope of this program are the cables used in sensitive instrumentation circuits with low level signals of the Nuclear Instrumentation systems (NIS) which includes Source Range Monitors (SRM), Intermediate Range Monitors (IRM), Local Power Range Monitors (LPRM), and Radiation Monitoring systems which includes Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors. *For the LPRM, SRM, and IRM systems,* the cables within the scope of this AMP are the cables between the detectors and main control room panel. For the radiation detectors within the scope of this program, the cables within the scope of this AMP are the cables between the detectors and the associated meters.	No	
2.Preventive Actions	This is a surveillance testing program and no actions are taken as part of this program to prevent or mitigate aging degradation.	This is a surveillance testing program and no actions are taken as part of this program to prevent or mitigate aging degradation.	No	
3.Parameters Monitored / Inspected	The parameters monitored are determined from the plant technical specifications and are specific to the instrumentation loop being calibrated, as documented in the surveillance test procedure.	**Nuclear Instrumentation System:** Local Power Range Monitoring: (LPRM) *In accordance with* NUREG 1801, calibration surveillance testing is being credited for the LPRM system. The full core LPRM calibration is performed per technical specification surveillance requirements. Per the implementing procedure, the LPRMs are verified to be within calibration. The acceptability *of the LPRM cables/detectors/connectors is verified through this calibration.* This calibration adjusts for loss in sensitivity of the circuit. Cable testing as recommended by ISG-15 is not being credited. As recommended by ISG-15, Exelon is committing to a once every 10 year review of LPRM calibration results for cable aging degradation. The first review will be performed prior to entering the period of extended operation. Source Range Monitoring: (SRM) For the SRM system, the cables between the preamplifier and detectors are subject to Current/Voltage (I/V) testing. The I/V test data is used to calculate the cable insulation resistance. The I/V testing results will be indicative of reduced insulation resistance. These tests verify the insulation resistance of the cables inside the drywell, along with the operability of the detectors and connectors. A surveillance test of the SRM monitors is *performed to verify the functionality of the* SRM (indicate counts per cycle within a certain range or have proper signal to noise ratio) during core alterations (refueling). This surveillance test verifies the integrity of the SRM cable system. Cable and surveillance testing as recommended by ISG-15 is being credited for the SRM system. Intermediate Range Monitoring: (IRM)	No Yes	1 2

Program Elements	XI.E2, Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits NUREG-1801 Description	(B.1.37) Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Dresden-Quad Cities Basis	GALL Exception	Comments
		For the IRM system, the cables inside the drywell are subject to Current/Voltage (I/V) testing. The I/V test data is used to calculate the cable insulation resistance. The I/V testing results will be indicative of reduced insulation resistance. These tests verify the insulation resistance of the cables inside the drywell, along with the operability of the detectors and connectors. A surveillance response test will be performed for the IRM monitors from the preamplifier to the control room chassis by injecting simulated inputs into the preamplifier. This surveillance test will verify the integrity of the IRM cables between the preamplifier and control room chassis. Cable and surveillance testing as recommended by ISG-15 is being credited for the IRM system.	Yes	2, 3
		Radiation Monitoring System:		
		Drywell High Range Radiation Monitoring: In accordance with NUREG 1801, calibration surveillance testing is being credited for the Drywell High Range Radiation Monitors. The calibration required by technical specification surveillances will verify that the cables maintain adequate insulation resistance integrity to perform their intended function. In this calibration, a calibrated source is used to expose the detector to gamma radiation field, and verify that acceptable readings are measured on the corresponding meter. Cable testing as recommended by ISG-15 is not being credited. As recommended by ISG-15, Exelon is committing to a once every 10 year review of the calibration results for cable aging degradation. The first review will be performed prior to entering the period of extended operation.	No	1
		Main Steam Line Radiation Monitoring: (MSLRM) In accordance with NUREG 1801, calibration surveillance testing is being credited for the entire MSLRM system. The calibration utilizes a source capable of producing photon energy in the range expected during normal and abnormal conditions. This check is performed with the entire system, including detectors, cables, and control room chassis, intact. This demonstrates that no detector or connecting cable degradation has occurred that could inhibit the system from performing its intended function. Cable testing as recommended by ISG-15 is not being credited. As recommended by ISG-15, Exelon is committing to a once every 10 year review of the calibration results for cable aging degradation. The first review will be performed prior to entering the period of extended operation.	No	1
		Steam Jet Air Ejector Radiation Monitoring. In accordance with NUREG 1801, calibration surveillance testing is credited		

Program Elements	XI.E2, Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits NUREG-1801 Description	(B.1.37) Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Dresden-Quad Cities Basis	GALL Exception	Comments
		for the entire SJAERM system. The calibration utilizes a source capable of producing photon energy in the range expected during normal and abnormal conditions. This check is performed with the entire system, including detectors, cables, and control room chassis, intact. This demonstrates that no detector or connecting cable degradation has occurred that could inhibit the system from performing its intended function. Cable testing as recommended by ISG-15 is not being credited. As recommended by ISG-15, Exelon is committing to a once every 10 year review of the calibration results for cable aging degradation. The first review will be performed prior to entering the period of extended operation.	No	1
4.Detection of Aging Effects	Calibration provides sufficient indication of the need for corrective actions by monitoring key parameters and providing trending data base on acceptance criteria related to instrumentation loop performance. The normal calibration frequency specified in the plant technical specification provides reasonable assurance that severe aging degradation will be detected prior to loss of the cable intended function. The first tests for license renewal are to be completed before the period of extended operation. Draft Interim Staff Guidance Letter ISG-15 states that cable testing and trending of calibration results are an acceptable aging management methods capable of detecting reduced insulation resistance before loss of the cable intended function.	The LPRM, Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors are calibrated per the frequency specified in the technical specification. The normal calibration frequency specified in the technical specification provides reasonable assurance that severe aging degradation will be detected prior to loss of the cable intended function. A review of calibration results will be completed before the period of extended operation and every 10 years thereafter. This review may detect severe aging degradation prior to the loss of cable intended function. The SRM and IRM cable systems inside the drywell are tested for insulation resistances. This test is a direct indication of condition of the insulation and will detect severe aging degradation prior to the loss of cable intended function. These cable systems are being tested every 24 months. The SRM surveillance test is performed every 24 months and will provides reasonable assurance that severe aging degradation will be detected prior to loss of the cable intended function. A review of the surveillance results will be completed before the end of the current term and every 10 years thereafter. This review may detect severe aging degradation prior to the loss of cable intended function. The IRM surveillance test will be performed before the period of extended operation and every 24 months thereafter. The surveillance test will provides reasonable assurance that severe aging degradation will be detected prior to loss of the cable intended function. A review of the surveillance results will be completed before the period of extended operation and every 10 years thereafter. This review may detect severe aging degradation prior to the loss of cable intended function.	No Yes Yes Yes	
5.Monitoring and Trending	Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement, tests results that are trendable provide additional information on the rate of degradation.	Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement of NUREG 1801, calibration results will be trended by the once every 10 year review of calibration test results, as recommended by ISG-15.	No	

	XI.E2: Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits	(B.1.37) Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits		
Program Elements	NUREG-1801 Description	Dresden-Quad Cities Basis	GALL Exception	Comments
6.Acceptance Criteria	Calibration readings are to be within the loop-specific acceptance criteria, as set out in the plant technical specifications surveillance test procedures.	The LPRM, Drywell High Range Radiation Monitors, Main Steam Line (MSL) Radiation Monitors, and the Steam Jet Air Ejector (SJAE) Radiation Monitors calibration results are to be within the acceptance criteria, as set out in the technical specifications surveillance calibration procedures. The Source Range Monitors (SRM), Intermediate Range Monitors (IRM) cable systems test results and surveillance results are to be within the acceptance criteria, as set out in the testing and surveillance procedures.	No Yes	
7.Corrective Actions	Corrective Actions such as recalibration and circuit trouble-shooting are implemented when an instrument loop is found to be out of calibration. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address corrective actions.	For the LPRM, Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors corrective actions such as recalibration and circuit trouble-shooting are implemented when calibration results do not meet the acceptance criteria. When the loop cannot be recalibrated to meet the technical specifications surveillance calibration acceptance requirements a condition report will be written and corrective action taken. This meets the requirements of 10 CFR Part 50, Appendix B. A condition report will be written and corrective action taken for the SRM and the IRM cable systems that fail to meet the acceptance criteria of the cable tests and surveillance testing. This meets the requirements of 10 CFR Part 50, Appendix B.	No Yes	1 2, 3
8.Confirmation Process	As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.	The requirements of 10 CFR Part 50, Appendix B addresses the confirmation process.	No	
9.Administrative Controls	As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.	The requirements of 10 CFR Part 50, Appendix B addresses administrative controls.	No	
10.Operating Experience	Operating experience has shown that a significant number of cable failures are identified though routine calibration testing. Changes in instrument calibration can be caused by degradation of the circuit cable and are one indication of potential electrical cable degradation.	This is a new aging management program and therefore there is no programmatic operating experience. However, plant experience shows that when and equipment cannot be brought into calibration or when cable system tests indicate unacceptable results, further reviews will identify if the problem is attributable to the instrument, connector or cabling.	No	

Conclusion:

The Gall requirements, augmented by proposed Interim Staff Guidance (ISG)-15 is met by the Dresden and Quad Cities aging management program (AMP).

Exception:

1. The GALL requires calibration tests be performed as part of the plant technical specifications and are specific to the instrumentation loop being calibrated. For the Source Range Monitors (SRM), Intermediate Range Monitors (IRM) cable systems cable and surveillance testing is used to detect aging for the cable prior to loss of intended function. Such cable and surveillance testing is acceptable in detecting aging degradation prior to the loss of cable intended function and is consistent with proposed interim staff guidance (ISG)-15.

Comments:

1. Procedures for the calibration of the LRPM, Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors require enhancement to include a discussion pertaining to a review of the calibration results for cable for aging degradation once every 10 years and clear guidance for the initiation of corrective action reports.
2. New procedures are required for the testing of the SRM and IRM cables at Quad Cities. Presently Quad Cities used predefines and work order instructions to test these cables. The new IRM procedure shall include the requirement to perform a surveillance response test of the IRM from the preamplifier to the control room chassis by injecting simulated inputs into the preamplifier, a discussion pertaining to a review of the surveillance results for cable for aging degradation before the period of extended operation and every 10 years thereafter, and clear guidance for the initiation of corrective action reports.
3. The Dresden IRM procedure needs to be enhanced to require a surveillance response test of the IRM from the preamplifier to the control room chassis by injecting simulated inputs into the preamplifier, a discussion pertaining to a review of the surveillance results for cable for aging degradation before the period of extended operation and every 10 years thereafter, and clear guidance for the initiation of corrective action reports.

RAI B.1.2 Supplemental Information Request

1) Provide the additional information for the corrosion performance of aluminum relative to carbon and stainless steel as outlined in the AMP B.1.2 Water Chemistry RAI response, especially in light of the statement made by the applicant that the Dresden aluminum tank bottoms have been replaced due to corrosion - what was the degradation mechanism, etc., and how will this be incorporated into inspection programs?

2) Regarding the one-time inspection for water chemistry the staff requests the applicant to provide an explanation regarding the one-time inspection of the SBLC system relative to crack initiation and SCC. The GALL report, Table VII E2, indicates that the appropriate AMP for stainless steel in SBLC is "Water Chemistry." - *[Information provided during audit]*

3) The staff noted that the applicant credits chemistry one-time inspections of carbon steel and stainless steel components for general, crevice and pitting corrosion. However the applicant indicated in their RAI response that they would be performing chemistry one-time inspections to detect only crevice corrosion. The staff requests the applicant to provide additional details regarding chemistry one-time inspections for detecting general corrosion and pitting corrosion.

4) The applicant's RAI response regarding Aluminum Tanks directs the staff to the Buried Piping and Tanks AMP (B.1.25). However, the RAI response to AMP B.1.25 indicates that Aluminum Tanks should have been included in Above Ground Carbon Steel Storage Tank AMP (B.1.20). The applicant needs to clarify where they intend to direct this RAI response.

Response

1) The response to RAI B.1.02, Item (h) contained the following statement:

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

Water Chemistry is credited with managing pitting and crevice corrosion for the in-scope aluminum storage tanks. Based on a review of EPRI 1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3, Appendix A (Treated Water), there is no appreciable difference in the corrosion resistance of aluminum compared to carbon steel or stainless steel for these two aging mechanisms. Therefore, the RAI response statement identified above should have read:

> "Given the excellent corrosion resistance of aluminum, the Dresden and Quad Cities Water Chemistry Program will adequately manage the aging of the aluminum storage tanks by maintaining low water impurities."

There was no definitive aging mechanism identified for degradation of the subject Dresden aluminum tank bottoms (see response to Supplemental RAI B.1.20). The Above Ground Carbon Steel Tanks Program (see response to Supplemental RAI B.1.20) includes a requirement for performance of a one-time internal UT of the bottom of the aluminum Condensate Storage Tank or Demineralized Water Storage Tank at Quad Cities and a periodic UT thickness inspection of the bottoms of the in-scope aluminum tanks at Dresden. The Dresden UT thickness inspections will be performed at a frequency not to exceed once every 10 years. These UT inspections will identify any loss of material due to any aging mechanism for the affected tanks. The program will also include a visual internal/external inspection of the in-scope tanks at both sites for pitting and crevice corrosion at a rate not to exceed once every 5 years.

2) Section VII.E2 of NUREG-1801 addresses aging management for the Standby Liquid Control System. For stainless steel components exposed to a sodium pentaborate environment, NUREG-1801 specifies crack initiation and growth/stress corrosion cracking as the applicable aging effect/mechanism and recommends Aging Management Program XI.M2, "Water Chemistry." Unlike other instances where NUREG-1801 specifies a one-time inspection to verify the effectiveness of the chemistry control program, NUREG-1801 is silent concerning one-time inspection of SBLC components.

    When analyzing components exposed to a sodium pentaborate environment for aging management at Dresden and Quad Cities, Exelon agreed with NUREG-1801 and credited "Water Chemistry" as the appropriate aging management program. However, in Section 3.3.1.2.3 of the LRA, Exelon did take exception to the Water Chemistry program for SBLC components. Specifically, Exelon credited the SBLC make-up water chemistry rather than the chemistry of the sodium pentaborate. The technical justification provided was that the sodium pentaborate maintained in the SBLC storage tank would mask most of the chemistry parameters that need to be monitored for stress corrosion cracking. Control of the make-up water chemistry would be more effective at managing stress corrosion cracking. Exelon credited the subject one-time inspection of SBLC components in Section 3.3.1.2.3 of the LRA. <u>Specifically, Exelon committed to perform and inspection of a Dresden SBLC pump discharge valve and a Quad Cities SBLC pump casing. The purpose of the one-time inspection is to verify the effectiveness of the Water Chemistry Program at mitigating stress corrosion cracking. Upon further review, Exelon believes that water chemistry found inside the SBLC pump discharge valve may not be representative of that found inside the SBLC tank. As such, Exelon will eliminate the inspection of the Dresden SBLC pump discharge valve and replace it with a UT inspection of a Dresden SBLC storage tank containing sodium pentaborate. The inspection will be performed on the side of the tank as close to the tank bottom as possible. Four UT measurements will be taken in each quadrant and the sample population will be expanded if stress corrosion cracking is detected.</u>

3) The one-time inspections of carbon steel and stainless steel components will look for general, crevice and pitting corrosion.

The last sentence in the second paragraph of Exelon's response to RAI B.1.02, Item (g), reads:

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

To provide further clarity, this sentence should have read:

"General corrosion is more prevalent in carbon steel; and pitting and crevice corrosion are more prevalent in stainless steel. Both types of materials will be inspected for general, pitting and crevice corrosion."

In the bulleted paragraphs that are part of Exelon's response to RAI B.1.02, Item (g), a special focus is provided with regard to the inspection points for crevice corrosion. This special focus was provided because the focus of RAI B.1.02, Item (g) is corrosion in areas of low flow, and crevice corrosion is most likely to occur in areas of low flow. However, the special focus on inspection points for crevice corrosion was not intended to imply that the one-time inspections would look only for crevice corrosion. The one-time inspections will look for all three – general, pitting and crevice corrosion.

4) The response to RAI B.1.02, Item (h) indicated that a requirement to perform a UT of the in-scope aluminum storage tanks was included in the Dresden and Quad Cities Buried Piping and Tanks Inspection Program (AMP B.1.25). However, the subject UT is in fact to be included in the Above Ground Carbon Steel Tanks Program (AMP B.1.20).

RAI B.1.2-1 Supplemental Information Request

Potential Open Item B.1.2-1 the staff requested that the applicant provide information regarding how aging degradation of the SBLC tank and piping up to the pump will be managed since sampling chemistry downstream of the tank and receipt inspection of the chemicals used in the tank will not provide adequate assurance that degradation is not occurring in this section of the system.

Response

The response to RAI B.1.2 was revised as a result of this additional information request. Please refer to the revised response to RAI B.1.2 (above) concerning the sample population of SBLC components chosen for one time inspection. The revisions to RAI B.1.2 are underlined.

RAI B.1.3 (b) Supplemental Information Request

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.

Response:

The studs were manganese phosphate coated as part of the original General Electric Purchase specification requirements. Four studs at Dresden Unit 2 were found to have cracking during refuel outages D2R11 (studs 47 and 70) and D2R15 (studs 52 and 81). These studs were replaced. A failure analysis was performed on one of the closure studs found cracked during refuel outage D2R11. The analysis concluded that the cracking was the result of stress corrosion cracking (mixed propagation modes) which initiated at the base of pits located in the thread roots. The probable cause of the stress corrosion cracking was determined to be exposure of the studs to oxygenated water during outages while in the tensioned condition. Stress corrosion cracking was identified as the aging mechanism for the stud indications found during Dresden Unit 2 refuel outage D2R15 as well. No other recordable indications have been identified on the Dresden or Quad Cities reactor head closure studs.

The reactive vessel studs remain in the reactor vessel flange during refueling activities and are exposed to water during reactor cavity flood up. The original root cause reported that small amounts of water collected in the area around the bolt threads following the reactor cavity drain. Water collected in these small areas can not be removed following a refueling outage. As part of vessel re-assembly, the reactor head closure studs are tensioned. The combination of oxygenated water and tension resulted in the stress corrosion cracking.

Due to the nature of the installation, the only possible corrective actions are stud inspections and replacement as necessary. As such, Dresden and Quad Cities have not implemented any additional corrective actions other than bolt inspections as described in Exelon aging management program B.1.3, which requires inspections for cracking and loss of material for reactor head closure studs. The Reactor Head Closure Stud Program for Dresden and Quad Cities provides these inspections in accordance with NUREG-1801 Program XI.M3 requirements and has been proven effective in detecting the aging effect (cracking and loss of material) associated with all applicable aging mechanisms.

RAI B.2.2-1 Supplemental Information Request

In response to RAI B.2.2-1, the applicant by a letter dated October 3, 2003, stated that there are non-segregated bus ducts within the scope of license renewal that are not normally energized. These are bus ducts connecting the diesel generator to the ESF busses and connecting safety related buses. They are included in section 2.5.1of the license renewal application. These are not normally energized and are energized only for technical specification surveillance or emergency activities. They are only energized for very short durations during normal plant operation and are located inside (Reactor/Turbine/Diesel Generator/HPCI) buildings where the environment is free from moisture, wind, and extreme ambient temperature differences. Therefore, thermal aging is not a concern for the bus duct insulators or sleeves. There are no other aging mechanisms applicable for these bus ducts. Periodic surveillance testing performed per technical specification verifies functionality of the bus ducts. Dresden and Quad Cities operating experience including experience from the non-segregated bus duct (Reserve Auxiliary Transformer to 4 KV Busses) inspections currently performed at Dresden and Quad Cities also confirm that no aging mechanisms apply for these bus ducts that would affect their intended function. Thermal cycling for bolted connections is a concern for these bus ducts. Additionally, humidity and moisture could be a problem for the deenergized bus ducts.

Response

The non-segregated bus ducts connecting the Emergency Diesel Generators (EDG) to the ESF buses are not normally energized. When energized with the EDG at full load, the non-segregated bus ducts are loaded to only 33 % of the design capacity. The temperature rise effects due to energizing the non-segregated bus ducts are approximately 10 °C. The non-segregated bus ducts are designed for a temperature rise of 65 °C. Additionally, the non-segregated bus ducts are only energized for approximately 2 hours per month during the monthly diesel generator surveillance tests. This conservatively equates to less than one year of operation over the 60-year life of the non-segregated bus ducts.

The non-segregated bus duct connecting the safety buses is a cross-tie connection (Dresden only) that is not normally energized. This non-segregated bus duct is only energized during the once per 24-month surveillance test. This conservatively equal to less than one year of operation over the 60-year life of the non-segregated bus duct. Therefore, the temperature rise effects due to energizing the non-segregated bus duct are negligible.

The non-energized non-segregated bus duct bus bars are tubular aluminum with bolted joint connectors that are torqued to 65 ft. lbs. Each joint connector is filled around the bolts/nuts with Duxseal and then taped to provide a smooth surface. The available drawings do not indicate the bolting material. Exelon believes based on discussion with the vendor that the bolts are zinc plated high strength steel or stainless steel. The vendor manual states that under normal operating conditions, no internal maintenance is required on the bus ducts. Additionally, EPRI TR104213 Section 8.2 states the bolts should be inspected for evidence of overheating, signs of burning or discoloration, and indications of loose bolts. The bolts should not be retorqued unless the joint requires service or the bolts are clearly loose. Exelon believes that there are no credible aging effects concerning bus duct bolted connections that require management. However, Exelon will include these bus ducts in the B.2.2 (Periodic Inspection of Non-EQ, Non-Segregated Electrical Bus Ducts) inspection program to inspect 10 % of the bus bar insulation splice material at the bolted connections for surface anomalies, such as embrittlement, discoloration, cracking, chipping, or surface contamination. The absence of insulation material surface anomalies, such as embrittlement, discoloration, cracking, chipping, and discoloration provides positive indication that the bolted connections are not loose and therefore, the intended function of the bus duct will be maintained during the period of extended operation. This inspection will verify that there are no insulation material surface anomalies, such as embrittlement, discoloration, cracking, chipping, and discoloration of the bus bar insulation splice material at the bolted connections. <u>The inspection will also include a verification for the presence of dirt and moisture in the bus duct. The visual inspection will include as much of the insulation as can be seen in both directions beyond the location of the bolted material.</u> The initial baseline inspections will be completed prior to the beginning of the period of extended operation. Follow-up inspections will be performed on a frequency not to exceed once every ten years. If degradation is found that could adversely effect the intended function of the bus bar, inspections will be expanded appropriately to determine the extent of condition.

As stated in the original response to RAI B.2.2-1, these bus ducts are located inside the Reactor, Turbine, Diesel Generator, and HPCI buildings where the environment is free from moisture, wind and extreme ambient temperature differences. Humidity and moisture is not a credible aging effect that required management.