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10 CFR 54

RS-04-057

April 9, 2004

U. S. Nuclear Regulatory Commission
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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: Response to License Renewal Safety Evaluation Report for the Dresden and Quad Cities Nuclear Power Stations

Reference: Letter from Pao-Tsin Kuo (U.S. NRC) to John Skolds (Exelon Generation Company, LLC), "License Renewal Safety Evaluation Report for the Dresden and Quad Cities Nuclear Power Stations," dated February 12, 2004

Exelon Generation Company, LLC (EGC) is submitting the response requested in the referenced letter. Attachment 1 is the responses to the Safety Evaluation Report Open and Confirmatory items. Attachment 2 is comments on the text of the Safety Evaluation Report. Attachment 3 is the General Electric Company Upper Shelf Energy Evaluation report.

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

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

Respectfully,

April 9, 2004
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing

Attachment 1: Response to SER Open and Confirmatory items
Attachment 2: Comments on the SER text
Attachment 3: General Electric Upper Shelf Energy Evaluation Report

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 – Division of Nuclear Safety

Attachment 1

Response to SER Open and Confirmatory items

## Response to Open Items

<u>OI-2.1-1:</u> (Section 2.1.3.1.2 - Application of the Scoping Criteria in 10 CFR 54.4(a)(2))

The applicant did not provide an adequate basis in its response to staff RAI 2.1-2. The staff determined that the applicant did not provide a sufficient basis for limiting consideration of fluid spray interactions to only those non-safety related SSCs located within 20 feet of an active safety related SSCs. The staff requires additional clarification regarding the capability of active and passive safety-related SSCs located greater than 20 feet from a potential spray source to tolerate wetting, the specific operating experience that was relied upon to determine that it was not credible for fluid sprays to affect equipment greater than 20 feet from a failure location, specific methods to detect leakage in normally accessible and inaccessible areas, and justification for use of exposure duration in limiting the scope of potential failure mechanisms considered during scoping.

Response

Exelon has revised the methodology utilized in the scoping of non-safety related moderate energy piping systems that have the potential to spatially interact with safety related systems. Specifically, Exelon has eliminated the 20 foot separation criterion previously utilized to exclude moderate energy systems from the scope of License Renewal. The revised methodology assumes that all safety related components, active as well as passive, could be adversely affected by spray or wetting from a non-safety moderate energy system located in the same general area of the plant. As such, early detection of leakage was also eliminated from the revised scoping methodology.

Under the revised scoping methodology, all components from moderate energy non-safety related systems located in the same general area as a safety related component (active or passive) will be included within the scope of license renewal. "General area" is defined as the same floor (elevation) of a major building with no barrier walls between the fluid source and the safety related component. Barrier walls were defined as barriers that form the boundary of a room on the same elevation of a major building separating the safety related components from a spray or leak generated by a non-safety related component located on the other side of the barrier wall. All barrier walls credited for protection of safety related components were previously included within the scope of license renewal during the scoping of structures and are included in the structures monitoring aging management program described in section B.1.30 of the license renewal application and the masonry wall aging management program described in section B.1.29 of the license renewal application.

Exelon has not yet completed all of the evaluations resulting from this methodology change for non-safety related piping systems that can spatially interact with safety related systems. However, the results of the evaluations completed thus far are included in the remainder of this response. When all evaluations are completed, Exelon will provide the NRC with an updated response that includes a list of all scoping changes and additional aging management activities associated with each.

Following the revised methodology, portions of non-safety related moderate energy systems previously excluded from the scope of license renewal will be added to the scope of license renewal. The non-safety related moderate energy systems previously included in the scope of license renewal whose boundaries were expanded are listed in Table 1 below.

Table 1
Non-Safety Related Moderate Energy Systems Previously Included Within the Scope of License Renewal With Boundary Expansions

| System | Site Affected | LRA Section |
|---|---|---|
| Service Water | Dresden and Quad Cities | 2.3.3.16 |
| Clean Demineralized Water Makeup System | Dresden | 2.3.3.18 |
| Condensate and Condensate Storage System | Dresden and Quad Cities | 2.3.4.3 |
| Reactor Building Closed Cooling Water | Dresden and Quad Cities | 2.3.3.17 |
| Fuel Pool Cooling and Filter Demineralizer | Dresden | 2.3.3.23 |
| Main Generator Hydrogen Seal Oil system | Quad Cities | 2.3.4.6 |
| Stator Water Cooling | Quad Cities | 2.3.4.7 |

As a result of the revised scoping methodology, three non-safety related moderate energy systems that had been previously excluded from the scope of license renewal have now been added to the scope. They are listed in Table 2 below. In each case, the non-safety related system had previously been excluded from the scope of license renewal because of the 20 foot separation criteria at one of the two sites. These three systems contain the same system and component intended functions as reported in Chapter 2 of the license renewal application.

Table 2
Non-Safety Related Moderate Energy System Added to the Scope of License Renewal

| System | Site Affected | LRA Section |
|---|---|---|
| Main Generator Hydrogen Seal Oil system | Dresden | 2.3.4.6 |
| Stator Water Cooling | Dresden | 2.3.4.7 |
| Fuel Pool Cooling and Filter Demineralizer | Quad Cities | 2.3.3.23 |

The boundary expansion of these systems in the scope of license renewal has resulted in some changes to the Dresden and Quad Cities License Renewal Application. These changes are included in this response below. Additions to LRA table line items are shown as **bolded** text, and removals from the table line items are shown as "strike-through" text. The entire tables are not repeated here; only those line items containing changes.

**Changes to the LRA Resulting From Scoping Methodology Changes**

1. Section 2.3.3.16, "Service Water System"

   a. LRA Table 2.3.3-16, "Component Groups Requiring Aging Management Review – Service Water System," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| **Strainer Bodies (spatial interaction) (Quad Cities only)** | **Leakage Boundary (spatial)** | **3.3.1.5, 3.3.1.15** |
| **Tubing (spatial interaction) ) (Quad Cities only** | **Leakage Boundary (spatial)** | **3.3.1.15, 3.3.2.40** |
| Valves (spatial interaction) | Leakage Boundary (spatial) | 3.3.1.15, 3.3.1.27, 3.3.2.23, **3.3.2.40**, 3.3.2.279, 3.3.2.280, 3.3.2.281, 3.3.2.300 |

2. Section 2.3.3.17, "Reactor Building Closed Cooling Water"

   a. LRA Table 2.3.3-17, "Component Groups Requiring Aging Management Review – Reactor Building Closed Cooling Water System," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Heat Exchangers (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.3.1.5, 3.3.2.77, 3.3.2.78 |
| Piping and Fittings (spatial interaction) ~~(Quad Cities only)~~ (Includes flow elements) | Leakage Boundary (spatial) | 3.3.1.5, 3.3.1.13, 3.3.2.40 |
| **Pumps (spatial interaction)** | **Leakage Boundary (spatial)** | **3.3.1.5, 3.3.1.13** |

3. Section 2.3.3.19, "Demineralized Water Makeup System"

   No changes to the LRA are required.

4. Section 2.3.3.23, "Fuel Pool Cooling and Filter Demineralizer System (In-Scope for Dresden Only)"

   a. "(In-Scope for Dresden Only)" is removed from the Section 2.3.3.23 heading.

   b. Quad Cities Station UFSAR Reference Section 9.1.3 is added under Sub-section "UFSAR References."

   c. New Quad Cities Station Boundary Diagrams LR-QDC-M-38 and LR-QDC-M-80 are added under Sub-section "License Renewal Boundary Diagram References."

d. The "(Dresden only)" qualification is removed from Sub-section "System Intended Functions," "Preclude adverse effects on safety-related SSCs."

e. LRA Table 2.3.3-23, "Component Groups Requiring Aging Management Review – Fuel Pool Cooling and Filter Demineralizer System," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Piping and Fittings (spatial interaction) ~~(Dresden only)~~ | Leakage Boundary (spatial) | 3.3.1.1, **3.3.1.5, 3.3.2.21,** 3.3.2.40, **3.3.2.143, 3.3.2.145, 3.3.2.302** |
| **Pumps (spatial interaction)** | **Leakage Boundary (spatial)** | **3.3.2.182, 3.3.2.300** |
| Sight Glasses (spatial interaction) ~~(Dresden only)~~ | Leakage Boundary (spatial) | 3.3.1.5, 3.3.2.198, 3.3.2.199 |
| Valves (spatial interaction) ~~(Dresden only)~~ | Leakage Boundary (spatial) | **3.3.1.1,** 3.3.1.5, **3.3.2.21, 3.3.2.40, 3.3.2.272,** 3.3.2.273, **3.3.2.314** |

5. Section 2.3.4.3, "Condensate and Condensate Storage Systems"

a. LRA Table 2.3.4-3, "Component Groups Requiring Aging Management Review – Condensate and Condensate Storage System," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Piping and Fittings (spatial interaction) ~~(Dresden only)~~ | Leakage Boundary (spatial) | 3.4.1.2, 3.4.1.3, 3.4.1.4 |
| **Pumps (spatial interaction)** | **Leakage Boundary (spatial)** | **3.4.1.2, 3.4.1.3** |
| **Valves (spatial interaction)** | **Leakage Boundary (spatial)** | **3.4.1.2, 3.4.1.3, 3.4.1.4** |

6. Section 2.3.4.6, "Turbine Oil system (In-Scope for Quad Cities Only)"

a. "(In-Scope for Quad Cities Only)" is removed from the Section 2.3.4.6 heading.

b. Under Sub-section "System Evaluation Boundary," the second paragraph is changed to read, "At Dresden and Quad Cities, portions of this system are in proximity to safety related electrical components."

c. Under Sub-section "UFSAR References," the Dresden Station UFSAR reference is changed from "Not applicable" to "None."

d. Under Sub-section "License Renewal Boundary Diagram References," new Dresden Station Boundary Diagrams LR-DRE-M-5350-1 and LR-DRE-M-5350-3 are added.

e. "(In-Scope for Quad Cities Only)" is removed from the Table 2.3.4-6 heading.

f. LRA Table 2.3.4-6, "Component Groups Requiring Aging Management Review - Turbine Oil System," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Closure Bolting ~~(Quad Cities only)~~ | Pressure Boundary | 3.4.1.6 |
| Filters/Strainers (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.1.3, 3.4.2.16 |
| Piping and Fittings (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.1.3, 3.4.2.32 |
| Pump Casings (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.9, 3.4.2.37 |
| Tanks (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.1.3, 3.4.2.43 |
| Valves (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.1.3, 3.4.2.50 |

7. Section 2.3.4.7, "Main Generator and Auxiliaries (In-Scope for Quad Cities Only)"

a. "(In-Scope for Quad Cities Only)" is removed from the Section 2.3.4.7 heading.

b. Under Sub-section "System Evaluation Boundary," the second paragraph is changed to read, "At Dresden and Quad Cities, portions of this system are in proximity to safety related electrical components."

c. Under Sub-section "UFSAR References," the Dresden Station UFSAR reference is changed from "Not applicable" to "UFSAR Section 8.3."

d. Under Sub-section "License Renewal Boundary Diagram References," new Dresden Station Boundary Diagrams LR-DRE-M-22A and LR-DRE-M-355A. are added.

e. "(In-Scope for Quad Cities Only)" is removed from the Table 2.3.4-7 heading.

f. LRA Table 2.3.4-7, "Component Groups Requiring Aging Management Review – Main Generator and Auxiliaries," is revised to read as follows:

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Heat Exchangers (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.20 |
| Housings (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.21 |
| Piping and Fittings (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.33 |
| Pumps (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.38 |
| Tanks (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.44 |

| Component | Component Intended Function | Aging Management Ref |
|---|---|---|
| Valves (spatial interaction) ~~(Quad Cities only)~~ | Leakage Boundary (spatial) | 3.4.2.11, 3.4.2.52 |

8. LRA Table 3.3-2, "Aging management review results for the auxiliary systems that are not addressed in NUREG- 1801" is revised to add the following line item:

| Ref No | Component Group | Material | Environment | Aging Effect/Mechanism | Aging Management Program | Discussion |
|---|---|---|---|---|---|---|
| 3.3.2.314 | Valves | Aluminum | <90°C (<194°F) treated water | Loss of material/ Pitting and crevice corrosion | Water Chemistry (B.1.2) | NUREG-1801 does not address aluminum in a reactor grade water environment. |

9. <u>Appendix B, Section B.2.7, "Generator Stator Water Chemistry Activities (Quad Cities Only)"</u>

   a. "Quad Cities Only" is removed from the Section B.2.7 heading.

10. Dresden Appendix A, Section A.2.7, "Not Used."

    a. Dresden Appendix A, Section A.2.7, is replaced with the following:

    <u>"A.2.7 Generator Stator Water Chemistry Activities</u>

    The generator stator water chemistry activities aging management program manages loss of material and cracking aging effects by monitoring and controlling water chemistry. Generator stator water chemistry control maintains high purity water in accordance with General Electric guidelines for stator cooling water systems. Generator stator water is continuously monitored for conductivity and an alarm annunciates if conductivity increases to a predetermined limit."

**OI-50-237/03-04-01, 50-249/03-04-01, 50-254/03-04-01, and 50-266/03-04-01:** (Section 2.1.3.1.2 - Application of the Scoping Criteria in 10 CFR 54.4(a)(2))

As documented in Regional Inspection Report 50-237/03-04(DRS), 50-249/03-04(DRS), 50-254/03-04(DRS), and 50-266/03-04(DRS), dated September 15, 2003, the inspectors questioned the applicant's definition of an equivalent anchor as used to determine the extent of nonsafety-related attached to safety-related systems that was included within the scope of the license renewal. Specifically, the applicant included non-safety related piping attached to safety-related pipe up to the point where the non-safety related piping was restrained in three orthogonal directions. In a letter dated October 20, 2003, the staff requested the applicant to clarify whether this methodology was consistent with the applicable plants CLB. Additionally, the staff requested justification that would demonstrate that failure of the nonsafety-related piping that was potentially excluded from the scope of license renewal would not adversely impact the safety-related portion of the piping system in accordance with 10 CFR 54.4(a)(2).

Response

Exelon provided the necessary information to resolve this open item in the response to RAI 2.1-2 supplemental information request. This response was contained in a transmittal letter to the NRC dated March 25, 2004.

**OI-3.5.2.3.2-1:** (Section 3.5.2.3.2- ASME Section XI, Subsection IWF (B.1.27))

The applicant's response to RAI B.1.27 did not address the staff's concern regarding the inspection of Class MC Supports. The applicant's existing IWF program is NOT consistent with GALL in that it does not include the inspection of Class MC supports. The staff requested additional information as detailed in the SER section cited above.

Response

The Exelon response to this open item is provided below. This response incorporates the response to Supplemental RAI B.1.27 submitted by letter dated March 25, 2004, but includes additional information subsequently requested by the NRC.

Response to Item (1)

Class MC components at Dresden and Quad Cities stations are divided into four groups based on the section of the containment in which they are installed. These groups are:

(a) Drywell
(b) Suppression chamber
(c) Vent system between the drywell and suppression chamber
(d) Piping that penetrates the primary containment.

The supports in each group are discussed separately below.

(a) Drywell

This group includes supports that provide structural support for the drywell portion of the primary containment. This group includes the following supports:

i. Drywell Steel Support Skirt and Anchor Bolts – The steel support member is part of the Class MC support, however, it is encased in concrete and is inaccessible, and is exempt from examination per ASME Section XI, IWF-1230 (components encased in concrete).

ii. Biological Shield to Containment Stabilizer – These supports are not currently inspected. However, prior to the end of the current term of operation the IWF program will be augmented to cover these Class MC supports.

iii. RPV Male Stabilizer Attached to Outside of Drywell Shell – This is a subset of the "Biological Shield to Containment Stabilizer" support.

iv. RPV Female Stabilizer and Anchor Rods (also referred to as Gib) Embedded in Reactor Building Concrete Wall – This is a subset of the "Biological Shield to Containment Stabilizer" support. A portion of this support is inaccessible and the inaccessible portion is exempt from examination per ASME Section XI, IWF-1230 (components encased in concrete).

(b) Suppression Chamber
This group contains supports that provide structural support for the suppression chamber (torus). Supports in this group include the following:

i. Suppression Chamber Ring Girder Vertical Supports and Base Plates - These supports are not currently inspected. However, prior to the end of the current term of operation the IWF program will be augmented to cover these Class MC supports.

ii. Suppression Chamber Saddle Supports and Base Plates - These supports are not currently inspected. However, prior to the end of the current term of operation the IWF program will be augmented to cover these Class MC supports.

iii. Suppression Chamber Seismic Restraints and Base Plates - These supports are not currently inspected. However, prior to the end of the current term of operation the IWF program will be augmented to cover these Class MC supports.

(c) Vent System Between Drywell and Suppression Chamber
This group contains supports that provide structural support on the vents that connect the drywell with the suppression pool.

i. Vent Header Vertical Column Supports - These supports are not currently inspected. However, prior to the end of the current term of operation, the IWF program will be augmented to cover these Class MC supports.

ii. Vent Header Downcomer Stiffener Plates – These supports are considered to be integral attachments (stiffeners) to a Class MC pressure retaining component. As such, they are presently included in the IWE program at each site.

iii. Vent Header Lateral Bracing – These supports are considered to be integral attachments (stiffeners) to a Class MC pressure retaining component. As such, they are presently included in the IWE program at each site.

iv. Vent Header Longitudinal Bracing – These supports are considered to be integral attachments (stiffeners) to a Class MC pressure retaining component. As such, they are presently included in the IWE program at each site.

The specific Class MC components listed above are also included in NUREG 1801, Volume 2, Chapter III, Section III B1.3 which recommends the IWF aging management program for the period of extended operation. Because the downcomer bracing was already included as part of the "ASME Section XI, Subsection IWE" aging management program (AMP), an exception was taken in the LRA to the "ASME Section XI, Subsection IWF" AMP. The impact of utilizing IWE instead of IWF (there was no impact) was provided in the response to RAI 3.5-14.

(d) <u>Class MC Piping Penetrating Primary Containment</u>
This group of piping supports at Dresden and Quad Cities stations include MC piping systems that penetrate and are attached to the primary containment. Examples of systems included in this category are: instrument air, service air, primary containment vent and purge piping, and reactor building closed cooling water. Component supports for this category of piping are not included in the ASME Section XI, Subsection IWF programs at either site.

The technical basis for this exclusion is found in Table IWF-2500-1 contained in Subsection IWF of ASME Section XI. Specifically, Item No. F1.40 of Table IWF-2500-1 only recommends the inspection of Class MC supports other than piping supports. The basis for excluding MC piping supports in Table IWF-2500-1 is found in IWF-1230 and IWE-1220.

Subsection IWF-1230 states:

> "Component supports exempt form the examination requirements of IWF-2000 are those connected to components and items exempted from examination under IWB-1220, IWC-1220, IWD-1220, and IWE-1220."

Subsection IWE-1220 states:

> "The following components (or parts of components) are exempted from the examination requirements of IWE-2000: (d) piping, pumps and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components shall be examined in accordance with the rules of IWB or IWC, as appropriate to the classification defined by the Design Specifications."

For the reasons stated above, Class MC piping supports at Dresden & Quad Cities stations are excluded from the ASME Section XI, Subsection IWF program.

The MC pipe supports at Dresden and Quad Cities stations are not included in the IWB or IWC programs at each site and are excluded from the IWE program. For the reasons stated above, they are exempt from the IWF program as specified in Subsections IWF-1220.

The MC pipe supports at Dresden and Quad Cities stations are managed for aging by visual inspections performed under the Structures Monitoring (SM) program, as described in the LRA, Appendix B, B.1.30,"Structures Monitoring Program." The SM program is intended to encompass all component supports, including Class MC, that are not included in the IWF program. As such, both programs are technically adequate to manage the aging effects of the component supports within their respective scopes. A comparison of the two programs, with respect to Class MC piping supports, is as follows:

- Both programs are based on sampling of the total support population. Once a sample is selected for inspection during the initial interval, this same sample is then inspected during successive intervals; 10 year intervals in the case of the IWF program, and five year intervals in the case of the SM program.

- ASME Subsection IWF, Table IWF-2500-1, addresses Class 1 (25% inspected each interval), Class 2 (15% inspected each interval), and Class 3 (10% inspected each interval) piping supports, but does not address Class MC piping supports. Class MC supports are addressed in Table IWF-2500-1 under "Supports Other Than Piping Supports (Class 1, 2, 3, and MC)," with 100% of the supports examined each inspection interval. However, Note (3) in Table IWF-2500-1 allows that for multiple components other than piping with a similar design, function, and service, the supports of only one of the multiple components are required to be examined.

- The SM program inspects a fixed number of supports every 5 years. These supports are selected as representative of the supports throughout the plant, including environmental conditions as well as configuration. The same supports in the selection are inspected every interval. A minimum of ten supports on Class MC piping will be included in the sample population for each unit, representing each environment – configuration combination that exists for systems that contain Class MC piping. A baseline inspection will be performed on the sample of ten MC supports prior to the period of extended operation. This task will be accomplished by adding the requirement for a sample of ten Class MC supports in the SM program sample population during the annual procedure update prior to December 31, 2004.

- The SM program inspects supports in a comprehensive fashion by plant area. Since the supports in any given plant area experience similar environments and all relevant support materials are covered by the program, any relevant age-related degradation potentially affecting MC supports will be detected and evaluated for extent of condition regardless of whether it occurs specifically on a Class MC support.

- The Exelon Structures Monitoring Program contains specific personnel qualifications for those administering the inspection program as well as those evaluating and performing the inspections.

  The qualifications of personnel administering an inspection program of structures are the following:
  - registered professional or structural engineer

- o Knowledgeable in the design, evaluation and performance requirements of nuclear structures.
- o Possess at least 5 years experience in structural and seismic engineering for nuclear structures
- o Be a degreed civil/structural engineer for an accredited college university.

The qualifications for personnel evaluating the results on an inspection are the following:

- o Knowledgeable in the design, evaluation, and performance requirements of nuclear structures.
- o Possess at least 5 years experience in structural and seismic engineering for nuclear structures.
- o Be a degreed civil/structural engineer from an accredited college or university.

The person performing the inspection of structures shall be suitably knowledgeable and/or trained to perform the activity. Personnel are suitably knowledgeable, thereby qualified via:

- o Demonstrating sufficient knowledge of inspection attributes through discussion with the administrator (or his designee) and/or
- o Performing an initial oversight of the individual's activities by the administrator (or his designee).

- The SM program presently includes component support inspection attributes for excessive deflection, distortion, misalignment, significant corrosion resulting in a loss of cross-section, loose bolting, cracked welds, and damaged grout pads. The SM program does not presently include the inspection of standard components such as snubbers, struts, and spring cans. But as stated in the response to RAI B.1.30 Supplemental Information Request, the SM program will be expanded to include the inspection of standard components such as snubbers, struts, and spring cans.

- The IWF program includes component support examination attributes for structural distortion or displacement of parts; loose, missing, cracked, or fractured parts, bolting or fasteners; corrosion or erosion that reduces cross-sectional area; misalignment of supports; improper hot or cold positions for snubbers and spring cans; and damaged or broken grout or concrete.

Response to Item (2)

Baseline inspections of typical samples of each type of Class MC component support added to the IWF program will be performed prior to the start of the period of extended operation.

Response to Item (3)

10 CFR 50.55a does not address Class MC component supports. ASME Section XI, Subsection IWF-1230 states:

"In addition, portions of supports that are inaccessible by being encased in concrete, buried underground, or encapsulated by guard pipe are also exempt from the examination requirements of IWF-2000."

Two of the Class MC component supports listed above in the response to Part (1) were identified as inaccessible due to being encased in concrete. They were the "Drywell Steel Support Skirt and Anchor Bolts" component support, and the Anchor Rods or Gib portions of the "RPV Female Stabilizer and Anchor Rods" component support. Loss of Material due to corrosion of the encased portion is not an aging effect requiring management. EPRI TR-114881, "Aging Effects for Structures and Structural Components (Structural Tools)," Section 5.3.1.5, states that, "The high alkalinity (pH > 12.5) of concrete provides an environment around embedded steel and steel reinforcement which protects them from corrosion." Therefore, the inaccessible portions of these supports do not require aging management.

Response to Item (4)

The response to RAI 2.4-2 Supplemental Information Request identifies items (b), (c), (d), and (j) as Class MC component supports with aging management references to the IWF program. Items (a), (e), (f), (g), (h), (i), and (k) are components other than Class MC component supports. With respect to Class MC component supports, the response to RAI 2.4-2 Supplemental Information Request is consistent with this response to RAI B.1.27 Supplemental Information Request.

Also, the response to RAI 2.4-10 is consistent with the response to RAI 2.4-2 Supplemental Information Request (and with this response to RAI B.1.27 Supplemental Information Request), with the following clarification. At the time the response to RAI 2.4-10, Item (b), was submitted, the phrase, "drywell lower ring support," listed in Part (b) of the request was assumed to be the drywell support skirt. Based on this, it was identified as a Class MC component support, being managed by the IWF program, the same as Item (j) in the response to RAI 2.4-2 Supplemental Information Request. On the other hand, if the requestor meant the 6x1-inch continuous steel ring on the interior bottom of the drywell discussed in the Quad Cities UFSAR Section 3.8.2 and UFSAR Figures 3.9-5 and 3.9-7, then it would be the same as RAI 2.4-2 Supplemental Information Request, Item (i). Since the interior shear ring is encased in concrete, as is the drywell steel support skirt, its aging management requirements are the same as previously discussed for the drywell steel support skirt.

**OI-4.2.1(c):** (Section 4.2.2.1 - limiting beltline materials USE values)

The applicant's response to RAI 4.2.1(c) did not address the staff's concern. The staff requested the applicant to provide all fluence data for all welds and plates in the beltline region and specify which one is bounding in determining the USE [upper shelf energy]. The staff needs to review the fluence data to evaluate the limiting beltline materials USE values presented in LRA tables 4.2.1-1 through 4.2.1-8. These are used for the determination of the bounding 54 effective full-power years (EFPY) 1/4T fluences for the D/QCNPS units.

Response

The Exelon response to this open item is provided in the response to Supplemental RAI 4.2-1(c), provided on page 26 of this attachment.

<u>OI-B.1.23-2:</u> (Section 3.0.3.10.2 - One Time Inspection (B.1.23) - Plant Heating System components)

The staff questioned the basis for using a one-time inspection in an environment that 1) varies with normal plant conditions, 2) is impractical to monitor or control routinely, and 3) is similar to the environments associated with the Aging Management References listed in part b of RAI B.1.23-2. The applicant responded that environments with these characteristics are air and steam; moist air; saturated air; warm moist air; moist containment atmosphere; steam or demineralized water; internal: occasional exposure to moist air; external; ambient plant air environment; dry gas; and hot diesel engine exhaust gases containing moisture and particulates. Based on the material and environment characteristics, the applicant believes that the aging effect is not expected to occur or is expected to progress slowly such that a one-time inspection is adequate to manage the aging effects. For carbon steel, cast iron, alloy steel, elastomers, and neoprene components in these environments, staff does not consider a one-time inspection adequate since aging effects are likely to occur in these material/environment combinations. Staff considers periodic inspections or a one-time inspection used to verify the adequacy of another AMP more appropriate to manage these components. The applicant is requested to provide additional information on the environmental conditions and the operating experience in order to justify the use of a one-time inspection, or provide periodic inspections for these components.

(Section 3.4.2.4.1 Main Steam System-One Time Inspection (B.1.23) - Plant Heating System components)

For the component NSR vents or drains, piping and valves addressed by AMR Reference 3.4.2.30, the applicant has identified that the material-environment includes carbon steel exposed to air, moisture, humidity, and leaking fluid. In its response to RAI B.1.23-2(b), the applicant implies that the loss of material due to corrosion is expected to be sufficiently slow that a one-time inspection can be used for aging management. The applicant has not provided sufficient information to justify the use of a one-time inspection. This is part of Open Item B.1.23-2.

Response

Exelon provided the necessary information to resolve open item OI-B.1.23-2 (Section 3.0.3.10.2) in the response to RAI B.1.23-2.6 supplemental information request included in the attachment to Exelon transmittal letter dated March 25, 2004. Exelon resolved open item OI-B.1.23-2 (Section 3.4.2.4.1) in the response to RAI B.1.23-2.2 supplemental information request included in the attachment to Exelon transmittal letter dated March 25, 2004.

Response to Confirmatory Items

CI.2.3.4.2-3: (Section 3.1.2.4.1 - AMR review of Reactor Vessel and internals)

The staff needs additional information from the applicant in order to evaluate the aging management of the capped CRD nozzles such as description of the configuration and location of the capped nozzle, description of how these welds and caps are managed etc. The applicant needs to include in the discussion the past inspection techniques applied, the results obtained, mitigative strategies followed, weld repairs carried out and any other relevant information.

Response

Exelon provided the necessary information to resolve this confirmatory item in the response to RAI 2.3.4.2-3 supplemental information request which was included in the attachment to Exelon transmittal letter dated January 26, 2004.

CI.3.0.3.14.2-1: (Section 3.0.3.14.2- Structures Monitoring Program (B.1.30) )

The additional information provided by the applicant in its response to RAI B.1.30 did not sufficiently address the questions posed by the staff. In order to completely address the questions in this RAI, the staff requests the applicant to confirm that: (a) the B.1.30 program covers non-ASME piping and components; and (b) there are no snubbers, struts and spring cans on non-ASME piping and components.

Response

Exelon provided the necessary information to resolve this confirmatory item in the response to RAI B.1.30 supplemental information request included in the attachment to Exelon transmittal letter dated December 5, 2003.

CI.3.1.2.3.2-1: (Section 3.1.2.3.2 - BWR Vessel ID Attachment Welds Program)

In response to the staff RAI 4.2-BWRVIPs, the applicant committed to perform a detailed review of the BWRVIP documents applicable to license renewal, prepare an amended response addressing items 1 through 7 for all of the documents applicable to license renewal, and submit it to the staff for review and approval.

Response

Exelon has completed the detailed review of BWRVIP documents applicable to license renewal and has prepared an amended response addressing items 1 through 7 for all of the documents applicable to license renewal. The amended response in provided below.

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

   The site-specific procedures at Dresden and Quad Cities implement all of the inspection methodologies contained in the applicable BWRVIP documents. Additionally the materials of and configurations at Dresden and Quad Cities are similar to those specified in the BWRVIP documents with one exception. The steam dryer hold-down bracket attachment weld described in Table 2-2 of BWRVIP-48 does not exist at Dresden Unit 3. Dresden Unit 3 is the same configuration as Dresden Unit 2 and Quad Cities Units 1 and 2.

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

   Dresden and Quad Cities commit to implementing the BWRVIP documents, with the exceptions as noted in attached Table 1.

3. Describe how the commitments will be tracked.

   All license renewal commitments are controlled by the Exelon commitment management process described in LS-AA-110, Commitment Management. Commitment tracking files will be generated for each individual activity credited to implement the requirements of the aging management program. Additionally, steps in site procedures that implement the various activities specified in the BWRVIP documents are annotated as NRC commitments and are referenced to commitment tracking files that contain sufficient documentation describing the source of the commitment. Commitments contained in Exelon procedures are additionally controlled by the Exelon commitment management process.

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

   The FSAR Supplement (LRA Appendix A) Programs A.1.1, A.1.2, A.1.4, A.1.8, A.1.9, and A.1.22 have been updated to reflect the applicable BWRVIP documents, and exceptions as noted in attached Table 1. A revised FSAR supplement incorporating these changes was submitted to the NRC in the attachment to Exelon transmittal letter dated March 5, 2004 as part of the annual update required by 10 CFR 54.21(b).

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

   The only Technical Specification change required for both sites involves revision to the site P-T Curves. The existing P-T curves will be revised for 54 EFPY prior to the extended term of operation.

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

   All applicable TLAAs are discussed in Section 4 of the LRA.

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

   Exelon has provided an amended response addressing items 1 through 6 for all BWRVIP documents applicable to license renewal.

**Table 1**
**Commitments and Exceptions to Applicable BWRVIP Documents**

| Number | Commitments, Exceptions, and Notes | *LRA Ref* |
|---|---|---|
| BWRVIP-05 | Dresden and Quad Cities will implement BWRVIP-05 "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" However, as noted in the NRC SERs for BWRVIP-05, dated 7/28/98 and 3/7/00 "...the results apply only for the initial 40-year license period of BWRs." Application for an extension of this relief for the 60-year period of extended operation was submitted to the NRC for review and approval on February 24, 2004. | *A.3.1.6* *A.3.1.7* |
| BWRVIP-18 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-18 "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines." The NRC SER for BWRVIP-18, dated 12/02/99 has one open item pertaining to uncertainties in measuring the flaw length by UT or VT methods when performing the flaw evaluation. Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the unresolved item. | *A.1.9* |
| BWRVIP-25 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-25 "BWR Core Plate Inspection and Flaw Evaluation Guidelines." The NRC SER for BWRVIP-25, dated 12/19/99 has one open item pertaining to inspection of Rim Hold-down bolts (that have not been structurally replaced by retaining wedges). As Dresden and Quad Cities have installed wedges, this open item is not applicable. However, Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the noted exception. | *A.1.9* |
| BWRVIP-26 | Dresden and Quad Cities will implement the guidance provided in BWRVIP-26 "BWR Top Guide Inspection and Flaw Evaluation Guidelines." Additionally, Dresden and Quad Cities will perform augmented inspections for the top guide similar to the inspections of control rod drive housing (CRDH) guide tubes. | *A.1.9* |
| BWRVIP-27 | Dresden and Quad Cities will implement the guidance provided in BWRVIP-27 "BWR Standby Liquid Control System/Core Plate $\Delta P$ Inspection and Flaw Evaluation Guidelines." The requirements of ASME Section XI will be implemented in accordance with 10 CFR 50.55(a). | *A.1.8* |
| BWRVIP-29 | Dresden and Quad Cities implement BWRVIP-79 "EPRI Report TR-103515-R2." in lieu of BWRVIP-29 "EPRI Report TR-103515-R1." NUREG 1801, Section XI AMP M.2, M.7, and M.8 recommend implementation of BWRVIP-29. However, this exception has been accepted by the NRC NRC during review of the LRA. | *N/A* |
| BWRVIP-38 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-38 "BWR Shroud Support Inspection an Flaw Evaluation Guidelines." The NRC SER for BWRVIP-38, dated 07/24/2000 has an open item pertaining to the need to inspect welds in the lower plenum (i.e. shroud support leg welds). This item is pending development of new techniques and *tooling. Dresden and Quad Cities will perform the additional inspections when new inspection techniques and tooling are* developed, incorporated into the applicable BWRVIP document(s), and approved by NRC SER. However, Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the noted exception. | *A.1.9* |
| BWRVIP-41 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-41 "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines." The NRC SER for BWRVIP-41, dated 02/13/2001 has one unresolved item pertaining to inspection of the inaccessible thermal sleeve welds. Dresden and Quad Cities will perform the additional inspections when new inspection techniques and tooling are developed, incorporated into the applicable BWRVIP document(s), and approved by NRC SER. Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the noted exception. | *A.1.9* |
| BWRVIP-42 | BWRVIP-42 "LPCI Coupling Inspection and Flaw Evaluation Guidelines" is not applicable. Dresden and Quad Cities do not have LPCI Couplings. | *N/A* |

| Number | Commitments, Exceptions, and Notes | *LRA Ref* |
|---|---|---|
| BWRVIP-47 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-47 "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines." The NRC SER for BWRVIP-47, dated 10/13/1999 has one unresolved item for the BWRVIP to address the issue of re-inspection in the future after initial baseline inspections have been completed by a majority of U. S. BWRs. Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the noted exception. | *A.1.9* |
| BWRVIP-48 | Dresden and Quad Cities will implement the guidance provided in BWRVIP-48 "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines." | *A.1.4* |
| BWRVIP-49 | Dresden and Quad Cities will implement the guidance provided in BWRVIP-49 "Instrument Penetration Inspection and Flaw Evaluation Guidelines." | *A.1.8* |
| BWRVIP-74 | Dresden and Quad Cities will implement the guidance of BWRVIP-74 "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines" with the following exception. Tech Spec revisions containing new P-T Curves will be submitted prior to the term of extended operation.<br>Exceptions:<br>• Dresden and Quad Cities implement Risk informed ISI to supplement the ISI and GL 88-01 programs<br>• Quad Cities implements a NRC approved code case for inspection of the Reactor Vessel Leak Detection Line. | *A.1.1* |
| BWRVIP-75 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-75 "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," with Exception – The Relief Request submitted for the implementation of RISI indicates the Category A Welds are "subsumed into the RISI program." The NRC SER for BWRVIP-75, dated 05/14/2002 has one unresolved item, which pertains to the use of FOI (Factors of Improvement) for BWR austenitic stainless steel piping. However, Exelon reserves the right to amend this commitment pending final NRC approval and resolution of the unresolved item. | *A.1.7* |
| BWRVIP-76 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-76 "BWR Core Shroud Inspection and Flaw Evaluation Guidelines." However, the NRC has not issued a Final Safety Evaluation approving BWRVIP-76. Exelon, reserves the right to amend this commitment pending final NRC approval. | *A.1.9* |
| BWRVIP-78 | BWRVIP-78 "BWR Integrated Surveillance Program Plan" has been superseded by BWRVIP-86-A | *A.1.22* |
| BWRVIP-79 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-79"EPRI Report TR-103515-R2." However, the NRC has not issued a Final Safety Evaluation approving BWRVIP-79. Exelon, reserves the right to amend this commitment pending final NRC approval. | *A.1.2* |
| BWRVIP-86 | Dresden and Quad Cities will implement BWRVIP-86 "BWR Integrated Surveillance Program Implementation Plan" prior to the period of extended operation. However, this report addresses current term (32 EFPY) only. Therefore, it does not apply to the extended term of operation. | *A.1.22* |
| BWRVIP-104 | Dresden and Quad Cities will implement the general guidance provided in BWRVIP-104 "Evaluation and Recommendations to Address Shroud support Cracking in BWRs." However, the NRC has not issued a Final Safety Evaluation approving BWRVIP-104. Exelon, reserves the right to amend this commitment pending final NRC approval. | *A.1.9* |
| BWRVIP-116 | Dresden and Quad Cities will implement BWRVIP-116 "Integrated Surveillance Program (ISP) Implementation for License Renewal" if approved by the NRC. If BWRVIP-116 is not approved, Exelon will provide a plant-specific surveillance plan for the LR period in accordance with 10CFR Part 50, Appendices G and H. | *A.1.22* |

CI.3.1.2.3.6-1: (Section 3.1.2.3.6 - BWR Vessel Internals Program)

The staff issued RAI B.1.9-b requesting the applicant to confirm whether D/QCNPS follows the BWRVIP-25 guidelines for managing aging of the rim hold-down bolts and, if so, then identify and evaluate whether the projected stress relaxation in the rim hold-down bolts is a TLAA. In response to RAI B.1.9-b however, the applicant did not specify whether stress relaxation in the rim hold down bolts is a TLAA. In response to the staffs follow-up question, the applicant stated that the stress relaxation of the rim hold-down bolts is not a TLAA for Dresden and Quad Cities since wedge retainers structurally replace the lateral load resistance provided by the rim hold-down bolts. This is a confirmatory item pending formal submittal from the applicant.

Response

Exelon provided the necessary information to resolve this confirmatory item in the response to RAI B.1.9-b supplemental information request included in the attachment to Exelon transmittal letter dated January 26, 2004.

CI.3.1.2.3.8-1: (Section 3.1.2.3.8 - Reactor Vessel Surveillance program)

In response to Part 2 of Supplemental RAI B.1.22, in a letter dated November 21, 2003, the applicant stated that if staff does not approve the proposed BWRVIP-116, the applicant will provide a plant-specific surveillance plan for the license renewal period in accordance with 10 CFR Part 50, Appendices G and H, prior to entering the renewed license period. This is Commitment #22 in Appendix A of this SER.

Response

Exelon concurs that if the NRC does not approve the proposed BWRVIP-116, Exelon will provide a plant specific surveillance plan for the license renewal period in accordance with 10 CFR Part 50, Appendices G and H, prior to entering the renewed license period. This commitment was included as commitment #22 in the commitment list submitted by letter dated February 3, 2004.

CI.3.1.2.4.2-1: (Section 3.1.2.4.2 Reactor Vessel Internals (Including Fuel Assemblies and Control Blades))

The response to RAI 3.1.7b states that Dresden and Quad Cities will implement the BWRVIP recommendations and manage the effects of aging of IASCC through AMPs B.1.2 (Water Chemistry) and B.1.9 (BWR Vessel Internals). AMP B.1.9 is consistent with NUREG-1801 which references the use of BWRVIP-26 for the inspection of the reactor vessel internals, including the top guide, and BWRVIP-76 for the inspection of the shroud. However, according to Table 2-1 of BWRVIP-76, when fluences exceed $5 \times 10^{20}$ $n/cm^2$, a plant-specific analysis is required to be submitted to the NRC. This is a confirmatory item pending applicant's submittal of this analysis to the staff.

Response

The purpose of Table 2-1 contained in BWRVIP-76, BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines is to provide guidance concerning core shroud inspection intervals for welds contained in un-repaired core shrouds. Note 4 of the Table 2-1 indicates that for plants where fluence at the shroud exceeds $5x10^{20}$ n/cm$^2$, a plant specific analysis is required to be submitted to the NRC. However, this analysis only applies to those licensees that have not installed repairs to the core shroud. Dresden and Quad Cities have installed repairs for the core shrouds on all four units that structurally replace the horizontal welds (reference NRC Safety Evaluation Regarding Core Shroud Repair dated December 6, 1995, TAC Nos. M91301, M91302, and M93584 for Dresden; NRC Safety Evaluation Regarding Core Shroud Repair dated June 8, 1995, TAC Nos. M91301 and M91302 for Quad Cities). Since these repairs are installed, the shroud inspection frequency is determined using the guidance contained in Section 3 of BWRVIP-76, Inspection Strategy for Welds in Repaired Shrouds. Exelon inspects the vertical core shroud welds in accordance with BWRVIP-76 Section 3. Any necessary flaw evaluations due to these inspections will be performed in accordance with BWRVIP-76 for the term of extended operation.

<u>CI.4.2.1:</u> (Section 4.2.2.1 - Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement)

The results presented in LRA Tables 4.2.1-1 through 4.2.1-8 show that the percent reductions in USE for limiting beltline plates and welds for all four D/QCNPS units are less than the BWRVIP-74 equivalent margin analysis acceptance criteria. For Quad Cities Unit 2 beltline weld material, the predicted value of 39 percent is equal to the generic value. Both of these USE values are predicted using RG 1.99, Position 2.2 which requires some interpolation, and thus can affect the USE values. This is a confirmatory item pending applicants submittal providing details on how these USE values are calculated, so that the staff can confirm the values in the applicants analysis.

Response

The Exelon response to this confirmatory item is provided in the response to Supplemental RAI 4.2-1(a), provided on page 26 of this attachment.

CI.4.2.1(a): (Section 4.2.2.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement)

In response to the staffs follow-up question to RAI 4.2.1, the applicant refers to an applicants letter dated July 31, 2003, regarding Additional Information Regarding Request for License Amendment for Pressure-Temperature Limits.≅ Figure 2 in this letter shows the pre-EPU and EPU peak axial flux distribution at the inside surface of the reactor pressure vessel. The pre-EPU and EPU axial flux distribution profiles are different, since the pre-EPU flux peaks at an elevation higher than the mid-plane, whereas the EPU flux peaks at the mid-plane. The applicant stated that for determining the peak 54-EFPY surface fluences at the lower shell material, lower shell welds and the lower to lower-intermediate shell girth weld, the axial flux distribution factor of 0.71 is applied for pre-EPU and 0.74 is applied for EPU conditions. The staff has independently verified the axial flux distribution factors using the data presented in the figure mentioned above and also verified the peak surface fluences for the lower shell and associated welds as calculated by the applicant. The staff finds the response acceptable because the applicant has used appropriate axial flux distribution factors for calculating the peak 54-EFPY surface fluence for the lower to lower-intermediate shell girth weld and all lower shell materials when determining the limiting bounding materials. This is a confirmatory item pending formal submittal from the applicant.

Response

The Exelon response to this confirmatory item is provided in the response to Supplemental RAI 4.2-1(b), provided on page 26 of this attachment.

CI.4.2.1.6: (Section 4.2.1.6 - Reactor Vessel Circumferential Weld Examination Relief)

The applicant is required to submit an update to LRA Section 4.2.6 to include the circumferential weld examination relief analysis for Quad Cities in accordance with 10 CFR 54.3(a) upon staff's approval of the May 16, 2003, relief request.

Response

Exelon provided an update to LRA Section 4.2.6 in an annual update of the LRA conducted in accordance with 10 CFR 54.21(b). The revision to Section 4.2.6 was included in the attachment to transmittal letter dated March 5, 2004.

CI.4.2.2: (Section 4.2.2.7 - Reactor Vessel Axial Weld Failure Probability)

In section 4.2.2.7 of the LRA, the applicant states that Dresden and Quad Cities have the same mean $RT_{NDT}$ because the initial $RT_{NDT}$ chemical composition, and 54-EFPY surface fluence are the same for the limiting beltline axial welds at Quad Cities and Dresden. A comparison of the mean RTNDT value of 91° C for the Clinton axial weld from Table 4.2-1 with the Dresden and Quad Cities value of 19° C (67° F) shows that the Clinton axial welds bounds the Dresden and Quad Cities welds. The applicant should confirm that Quad Cities, Units 1 and 2 have a mean RTNDT value of 19° C (67° F) and address this TLAA of the axial welds for Quad Cities in the UFSAR Supplement.

Response

Exelon provided an update to LRA Section 4.2.7 in an annual update of the LRA conducted in accordance with 10 CFR 54.21(b). This update confirmed the Quad Cities Units 1 and 2 mean $RT_{NDT}$ value) and addressed the TLAA of the axial welds for Quad Cities. The revision to Section 4.2.7 was included in the attachment to transmittal letter dated March 5, 2004. A revision to the UFSAR Supplement was also provided. Note that the correct Clinton value for mean $RT_{NDT}$ is 91° F, not 91° C.

<u>CI.B.1.2-1:</u> (Section 3.0.3.2 -Water Chemistry Program (B.1.2)

The applicant committed to perform an inspection of a Dresden SBLC pump discharge valve and a Quad Cities SBLC pump casing.

Response

Exelon provided the necessary information to resolve this confirmatory item in the response to Part 2 of RAI B.1.2 supplemental information request included in the attachment to Exelon transmittal letter dated December 22, 2003. Additional information was provided in the response to Part 2 of RAI B.1.2-1a supplemental information request included in the attachment to Exelon transmittal letter dated March 25, 2004.

<u>CI.B.1.17:</u> (Section 3.3.2.3.2 - AMP on BWR Reactor Water Cleanup System)

In the LRA, the applicant stated that the inspection of reactor water cleanup system (RWCU) piping is not required because RWCU system piping was replaced with IGSCC resistant piping in accordance with NRC GL 88-01. This was verified during an Aging Management Program audit conducted by the staff on October 7-8, 2003. This is a confirmatory item pending issuance of the Audit Report.

Response

No action is required by Exelon concerning this confirmatory item.

<u>CI.B.1.23-1:</u> (Section 3.0.3.10 - One Time Inspection (B.1.23)

The applicant will expand the scope of Aging Management Program B.2.5 (Lubricating Oil Monitoring) or Aging Management Program B.1.21 (Fuel Oil Chemistry) to include components in the Reactor Core Isolation Cooling (RCIC) System, the High Pressure Coolant Injection (HPCI) System, the Emergency Diesel Generator and Auxiliaries System, and the Station Blackout Diesel System that are exposed to an environment of lubricating oil or fuel oil. The One-Time Inspection Program will be used to verify the effectiveness of these aging management programs. Additional description of the CI.B.1.23-1 is provided below:

- The applicant developed AMP B.2.8, Periodic Inspection of Plant Heating System, to perform periodic inspections of selected plant heating system components that are exposed to an environment of saturated steam and condensate. The One-Time

Inspection Program is no longer credited to manage aging effects for these components since periodic inspections will be performed.

- In response to RAI 3.2.1.4-3, the applicant stated that hardening and loss of strength due to elastomer degradation in the flexible hoses in a containment nitrogen environment would be managed by the One-Time Inspection Program. Upon further review, the applicant believes that these hoses are made of stainless steel with an overall stainless steel outer braided jacket and are not comprised of an elastomer. The One-Time Inspection Program will be used to verify that the hoses are constructed of metal rather than an elastomer material. Based on this inspection, any elastomer hoses will be replaced with metal flexible hoses. If metal hoses are found to be installed, the One-Time Inspection Program will perform inspections for mechanical damage. (Section 3.4.2.4.1 - Main Steam System)

- For non-safety related (NSR) vents or drains, piping, and valves in the main control room system, shutdown cooling system, and control rod drive hydraulic system, the LRA identifies loss of material due to corrosion for carbon steel, stainless steel, brass, or bronze in an environment of air, moisture, humidity, and leaking fluid. The staff requested the applicant to describe the types of corrosion expected and to provide criteria for selecting one-time sample locations for these types of corrosion. The applicant clarified in RAI 3.3-2 response that general, crevice, and pitting corrosion are expected in these components. The applicant compiled a list of the in-scope NSR vents and drains for the various systems throughout the plants. The One-Time Inspection program will inspect a selected number of NSR vent and drains for the affected systems. The sample population will be representative of all material and environment combinations but may not include components for every system. The criteria used for selection of susceptible inspection locations are as follows: 1) Corrosiveness of fluid passing through the vent, drain, or piping when in service - those components servicing more corrosive fluids are given preference; 2) Duration of service when performing venting and draining operations - those components with higher durations of service are given preference; 3) Frequency of performance of venting and draining operations through the selected components - those components with higher performance frequencies are given preference; and 4) Period that component has been in service - those components that have been in service longest are given preference. In addition, the applicant stated that NSR vents and drains are attached to normally closed isolation valves and are not likely to contain moisture. Any appreciable leakage or condensation inside these vents and drains would be identified in the course of periodic operations or through the daily monitoring of unidentified inputs to radwaste. Malfunctioning isolation valves or other degraded conditions would be promptly repaired, replaced, or corrected. For the reasons stated above, the rate of material loss due to corrosion is expected to be slow.

- The applicants AMP B.2.5, Lubricating Oil Monitoring Activities, will be expanded to include the analysis of the turbine oil systems components (Dresden only) exposed to generator hydrogen seal oil and the main turbine and auxiliaries components exposed to turbine EHC fluid. The One-Time Inspection Program will be used to verify the effectiveness of this AMP. The staffs review of the Lubricating Oil Monitoring Activities program and the One-Time Inspection program are addressed in SER Sections 3.0.3.16 and 3.0.3.10, respectively. (Section 3.4.2.4.5 - Main Turbine and Auxiliary Systems, and Section 3.4.2.4.6 - Turbine Oil System (Quad Cities Only))

This is a confirmatory item pending formal submittal from the applicant.

Response

Exelon provided the appropriate revisions to aging management program B.2.5 Lubricating Oil Monitoring Activities in the response to RAI B.1.23.2.3 & RAI B.1.23.4 supplemental information requests which were included in the attachment to Exelon transmittal letter dated January 26, 2004.

Exelon provided a copy of aging management program B.2.8, Periodic Inspection of Plant Heating System in the response to Part a of RAI B.1.23 supplemental information request which was included in the attachment to Exelon transmittal letter dated March 25, 2004. Additional information was provided in the response to RAI B.1.23.2.5 supplemental information request which was included in the attachment to Exelon transmittal letter dated January 26, 2004.

The Exelon response concerning elastomer degradation in flexible hoses was addressed in the response to RAI B.1.23.2.1 supplemental information request which was included in the attachment to Exelon transmittal letter dated January 26, 2004.

The Exelon response concerning aging management of non-safety related vents and drains was contained in the response to RAI B.1.23.2.2 supplemental information requests which was included in the attachment to Exelon transmittal letter dated January 26, 2004.

<u>CI.B.1.23-2.5:</u> (Section 3.3.2.3.7 Periodic Inspection of Plant Heating System)

The applicant developed this program in response to staff questions regarding the use of the One-time Inspection program on the plant heating system. The program is not based on a GALL Report program; therefore, the applicant summarized the program in terms of the 10-element program as described in Branch Technical Position, Appendix A of the SRP-LR. The program will use periodic visual inspections for cracking, loss of material, or other evidence of aging to monitor the condition of the system. This is a confirmatory item pending formal submittal from the applicant.

Response

Exelon provided the necessary information to resolve this confirmatory item in the response to RAI B.1.23.2.5 supplemental information request included in the attachment to Exelon transmittal letter dated January 26, 2004.

<u>CI.B.1.25-1:</u> (Section 3.0.3.12 - Buried Piping and Tanks Inspection)

The staff requested additional clarifying information to confirm that the soil environment is not aggressive to buried concrete piping and to confirm whether all buried carbon steel piping is coated.

Response

Exelon provided the necessary information to resolve that portion of the confirmatory item associated with soil environment in the response to part 3 of RAI B.1.25 supplemental information request which was included in the attachment to Exelon transmittal letter dated December 12, 2003. Exelon provided the necessary information to resolve the remaining portion of the confirmatory item associated with coating for carbon steel piping in the response to RAI B.1.25-1 supplemental information request which was included in the attachment to Exelon transmittal letter dated March 25, 2004.

CI-B.2.5-1: (Section 3.0.3.16 Lube Oil Monitoring Activities)

In its October 3, 2003, response to RAI B.1.23-2(a), the applicant committed to include the following additional components in the scope of this program: components in the reactor core isolation cooling (RCIC) system, additional components in the high pressure coolant injection (HPCI) system, additional components in the emergency diesel generator and auxiliaries system, and additional components in the station blackout diesel system. In addition, the applicant committed to add components exposed to EHC oil (main turbine and auxiliary systems) and generator hydrogen seal oil (turbine oil system - Quad Cities only) to the scope of this program. The staff finds that adding the above components to the scope of this program is appropriate, since maintaining oil quality is important for preventing aging effects. However, the applicant has not provided updates to the program elements to address the increased scope of the program. The applicant is requested to provide the appropriate revisions to the 10 elements and the UFSAR summary description of this program.

Response

Exelon provided the appropriate revisions to aging management program B.2.5 Lubricating Oil Monitoring Activities in the response to RAI B.1.23.2.3 & 4 supplemental information requests which were included in the attachment to Exelon transmittal letter dated January 26, 2004.

RAI 4.2.1 Supplemental Information Request (Note: This RAI includes Open Item 4.2.1(c) and Confirmatory Item 4.2.1 and 4.2.1(a))

1) Tables 4.2.1-1 through 4.2.1-8 provided the ART and USE for the limiting base material and weld metal along with the surveillance capsule data. However, there seems to be a discrepancy between the %Drop in the Upper Shelf Energy (USE) provided in the LRA for the ESW weld metal, and the % Drop in USE previously submitted to the NRC (see table below). Please discuss this discrepancy and provide the analysis that demonstrates the % Drop in USE was equal to 39%, at the end of the extended license (F=3.9 x $10^{17}$ n/cm$^2$ at 1/4T in Table 4.2.1-8 of the LRA) when determined per position 2.2 of RG 1.99, Rev.2. Also, identify the references for the surveillance results presented in LRA Tables 4.2.1-1 to 4.2.1-8.

SURVEILLANCE CAPSULE % DROP IN USE

| Plant | Report No. or Letter No. | Capsule Fluence | Material | Initial USE (from Report) | Irrad. USE (from Report) | %Drop USE (from Report) | LRA %Drop USE |
|---|---|---|---|---|---|---|---|
| Dresden 2 | Table 13 to ComEd Response to Generic Letter 92-01, dated 7/1/92 forward by ComEd letter dated 12/6/94 | 5.2 E16 | ESW | 139 | 133 | 4 | NA |
| Dresden 3 | same as above | 7.1 E16 | ESW | 72 | 72 | 0 | 0 |
| Quad Cities 1 | same as above and Report SwRI-06-7857, dated 8/84 | 5.5 E16 | ESW | 105 | 102 | 3 | 12 |
| Quad Cities 2 | same as above and Report SwRI-06-7484-002, dated 3/84 | 6.6 E16 | ESW | 125 | 90 | 28 | 32 |

(2) In response to RAI 4.2.1(a), the applicant states that it applied the axial flux distribution factor of 0.71 for calculating the peak pre-EPU fluence for the lower-to-lower intermediate shell girth weld and all lower shell materials. Describe how the pre-EPU axial flux profile compares with the post-EPU profile. Submit information about the axial flux distribution factor used for calculating the peak-EPU fluence for the lower-to-lower intermediate shell girth weld and all lower shell materials. [Confirmatory Item 4.2.1(a)]

(3) In RAI 4.2.1(c), the staff requested the applicant to provide all fluence data for all welds and plates in the beltline and specify the one that is bounding in determining the USE. In response to RAI 4.2.1(c) in a letter dated October 6, 2003, the applicant provides 54-EFPY surface fluences and 54 EFPY 1/4T fluences for all the beltline material, and identifies materials that are bounding in determining 54 EFPY ART, but does not identify the bounding materials for the USE. Identify the materials that are bounding in determining 54 EFPY USE. This is Open Item 4.2.1(c).

Confirmatory Item 4.2.1 (Draft SER Section 4.2.2.1)
The results presented in LRA Tables 4.2.1-1 through 4.2.1-8 show that the percent reductions in USE for limiting beltline plates and welds for all four D/QCNPS units are less than the BWRVIP-74 equivalent margin analysis acceptance criteria. For Quad Cities Unit 2 beltline weld material, the predicted value of 39 percent is equal to the generic value. Both of these USE values are predicted using RG 1.99, Position 2.2 which requires some interpolation, and thus can affect the USE values. This is a confirmatory item pending applicants submittal providing details on how these USE values are calculated, so that the staff can confirm the values in the applicants analysis.

Response

1) Response to Question 1 and Confirmatory Item 4.2.1:

Exelon has verified that the Dresden 2 values contained in supplemental information request above were incorrect for ESW and should have been reported as: Initial USE, 96 ft-lbs; Irradiated USE, 90 ft-lbs; and %Drop USE, 6%. The USE values of 139 and 133 ft-lbs apply to the plate material. The NRC subsequently requested confirmation of the Dresden 2 equivalent margin analysis for plate material.

General Electric and Exelon reviewed all of the original capsule evaluation reports in order to determine the basis for all reported upper shelf energy (USE) values. The Southwest Research Institute (SwRI) capsule reports determined USE values consistent with the definition of ASTM E 185-82 (i.e., averaging points with >95% shear). In addition, SwRI applied the same methodology to the unirradiated data in certain cases. This methodology resulted in discrepancies with original reported USE values for unirradiated material. However, the use of ASTM E 185-82 is an appropriate method for determining USE values.

In some cases the values reported in the LRA, though based on the capsule reports, did not follow the defined ASTM E 185-82 methodology. These cases are noted and re-evaluated in the following response. The results of this re-evaluation are presented for each unit, along with a conclusion concerning the impact on the projected 54 EFPY USE values provided in section 4.2.2 of the license renewal application (LRA). The LRA reported values will be changed where necessary in order to enforce consistency of approach.

**Dresden 2 Plate**
For Dresden 2, the plate material testing reported by General Electric in 1975 (Reference 9) reported a value of 153 ft-lbs for unirradiated USE. This value was also used in the LRA analysis. The 153 ft-lbs was obtained from a curve fit of the Charpy data points. The ComEd response to Generic Letter 92-01 (Reference 2) cites an unirradiated USE of 139 ft-lbs. This value was based on a Southwest Research Institute 1983 report (Reference 1) that considered the results of the first wall capsule as representing an unirradiated condition.

The 1983 SwRI report determined a base metal USE value of 133 ft-lbs for the second wall capsule. This value was based on a curve fit, The value of 139 ft-lbs for the first wall capsule originated in GE Report NEDC-12585, dated May 1975 (Reference 10), and was also based on a curve fit.

The data in Reference 9 indicates Charpy values of 143.9, 131.2, 147, and 166.8 ft-lbs for points with $\geq$ 95% shear. Averaging these data points in accordance with ASTM E 185-82 gives a value for unirradiated USE of 147.2 ft-lbs for the plate material.

The data in Reference 10 indicates Charpy data points of 140.8, 133, and 132.5 ft-lbs with $\geq$ 95% shear. Averaging these data points in accordance with ASTM E 185-82 gives a value for irradiated USE of 135.4 ft-lbs for the plate material in the first wall capsule.

The data in Reference 1indicates Charpy data points of 130.5, 125, and 143 ft-lbs with $\geq$ 95% shear. Averaging these data points in accordance with ASTM E 185-82 gives a value for irradiated USE of 132.8 ft-lbs for the plate material in the second wall capsule.

The values obtained with ASTM E 185-82 methodology were used to recalculate the EMA for Dresden 2 plate material. The % drop for the first wall capsule remains limiting. However, use of these values rather than the curve fit data previously reported changes the results, as indicated in revised LRA Table 4.2.1-1.

**Table 4.2.1-1: Equivalent Margin Analysis for Dresden Unit 2 Plate Material**

| BWR/3-6 PLATE |
|---|
| Surveillance Plate USE:<br>%Cu = 0.19<br>1st Capsule Fluence = $1.3 \times 10^{16}$ n/cm2<br>2nd Capsule Fluence = $5.2 \times 10^{16}$ n/cm2<br><br>1st Capsule Measured % Decrease = 8 (Charpy Curves)<br>2nd Capsule Measured % Decrease = 10 (Charpy Curves)<br><br>1st Capsule R.G. 1.99 Predicted % Decrease = 6 (R.G. 1.99, Figure 2)<br>2nd Capsule R.G. 1.99 Predicted % Decrease = 8 (R.G. 1.99, Figure 2) |
| Limiting Beltline Plate USE:<br>%Cu = 0.23<br>54 EFPY 1/4T Fluence = $3.9 \times 10^{17}$ n/cm2<br>R.G. 1.99 Predicted % Decrease = 15.5 (R.G. 1.99, Figure 2)<br>Adjusted % Decrease = 17.5 (R.G. 1.99, Position 2.2) |
| 17.5 < 23.5%, so vessel plates are bounded by equivalent margin analysis |

**Dresden 2 Weld**

Exelon does not have any record of unirradiated specimen test results for electroslag welds (ESW) for the Dresden Unit 2 reactor vessel to determine a baseline Upper Shelf Energy (USE) value. The "N/A" reported in the LRA reflects the lack of unirradiated data for Dresden 2. In the absence of baseline data, an assumption was made by Southwest Research Institute (SwRI) in 1983 (Reference 1) that the test results from the 35° wall capsule removed from the vessel after two years of operation were representative of the unirradiated condition. The basis for this assumption was the low fluence (1.3E16 n/$cm^2$) received by these specimens. The unirradiated USE value of 96 ft-lbs reported to the NRC in 1992 in response to Generic Letter 92-01 (Reference 2) was based on the testing results of this first near-wall capsule. The 96 ft-lbs USE value was taken from Table 15 and the curve plot in Figure 1B of Reference 10.

The 1983 SwRI report determined an irradiated value for USE of 90 ft-lbs. The data indicates two points on the upper shelf had greater than 95% shear, and these points had an average impact energy of 89.5 ft-lbs. The value of 96 ft-lbs was used as the unirradiated USE value, resulting in the 6% decrease in USE reported in Reference 2.

Review of the Charpy data in NEDC-12585 indicates upper shelf data points of 111.1 and 78.4 ft-lbs. Averaging these data points results in 94.5 ft-lbs for the USE value. Use of 94.5 ft-lbs as the unirradiated USE has no impact on the final results presented in LRA Table 4.2.1-2, since the R.G. 1.99 predicted value of 9% decrease in USE for the second near-wall capsule was used in the LRA evaluation of limiting beltline weld USE. This 9% decrease also bounds the 6% decrease obtained by using 96 ft-lbs as the unirradiated USE.

**Dresden 3**

For Dresden 3, the ESW material testing reported by Battelle Columbus Laboratories in 1975 (Reference 4) reported a value of 70 ft-lbs for unirradiated USE. This value was also used in the LRA analysis. The ComEd response to Generic Letter 92-01 (Reference 2) cites an unirradiated USE of 72 ft-lbs. This number was based on an average of the two data points with >95% shear for the unirradiated ESW material; the average of these two points gives a value for USE of 71.5 ft-lbs.

A 1984 Southwest Research Institute report presented test results from the 3rd (215°) Unit 3 reactor vessel capsule (Reference 3). In this report, Southwest Research cited a USE drop of "nil" (Reference 3). According to this test report, the single data point of >95% shear from the third capsule had an impact energy reading of 72 ft-lbs.

The zero %decrease was reported in the LRA analysis, though the reported initial USE of 70 ft-lbs differed from the 72 ft-lbs obtained with the ASTM E 185-82 methodology. Using the value of 72 ft-lbs as the unirradiated USE has no impact on the final results presented in LRA Table 4.2.1-4, since the R.G. 1.99 predicted decrease of 11% of was used in the evaluation rather than the measured % decrease of zero for the third near-wall capsule. The limiting beltline weld USE remains as reported in Table 4.2.1-4 of the LRA.

**Quad Cities 1**

For Quad Cities 1, the LRA evaluation used an unirradiated USE of >100 ft-lbs as reported in the "Results and Discussion" section of *the original unirradiated material testing* by Battelle Columbus Laboratories in 1975 (Reference 6). The ComEd response to Generic Letter 92-01 in 1992 (Reference 2) cites an unirradiated USE of 105 ft-lbs. This value was based on the curve fit for the unirradiated data points as plotted in Figure 5 of a 1984 report published by the Southwest Research Institute (Reference 5). The SwRI report presented the results from capsule 8, removed from the reactor in 1982. Table V of the SwRI document reported the unirradiated Charpy impact energy as 108 ft-lbs with 100% shear. Table VI of the 1984 SwRI report states that the ESW material test resulted in a 3 ft-lb decrease in USE. This value was derived from the single data point at 108 ft-lbs with > 95% shear in the Battelle 1975 report (Reference 6) for the unirradiated sample and another single data point at 105 ft-lbs with > 95% shear for the irradiated sample. This 3 ft-lb decrease then became the source for the irradiated USE value of 102 ft-lbs and 3% decrease in USE reported by ComEd in response to Generic Letter 92-01 (Reference 2).

The LRA evaluation obtained an irradiated USE of 88 ft-lbs from the hyperbolic tangent curve plotted in Figure 5 of the 1984 SwRI report (Reference 5), and is therefore not consistent with the ASTM E 185-82 definition for USE (average of points >95% shear). The LRA calculation using 100 ft-lbs unirradiated USE and 88 ft-lbs irradiated USE (12% drop) results in a projected decrease of 17.5% (less than 39%) for the 54 EFPY USE. This evaluation bounds the projected decrease from the 3 ft-lb (3%) drop measured in the 1984 Southwest Research Institute report and subsequently reported in the ComEd response to Generic Letter 92-01 (References 5 and 2).

The R.G. 1.99 predicted %decrease of 18.5% for 54 EFPY bounds the value determined from the LRA analysis. The data set used for the LRA evaluation of USE has no impact on the qualification of the Quad Cities 1 ESW material for 54 EFPY USE since the bounding R.G. 1.99 predicted value for USE decrease results in a final %decrease <39%.

**Quad Cities 2**

The original Quad Cities 2 value of 125 ft-lbs for unirradiated USE was reported in the 1975 Battelle Columbus Laboratories report on unirradiated properties for both of the Quad Cities units (Reference 6). In response to Generic Letter 92-01 (Reference 2), ComEd again reported for Quad Cities 2 an unirradiated USE of 125 ft-lbs.

In calculating the %decrease for the Quad Cities Unit 2 ESW, Exelon used a value of 121 ft-lbs for unirradiated USE in the LRA. This value was obtained from the curve fit provided in Figure 5 of a 1984 Southwest Research Institute report concerning the results from Quad Cities Unit 2 reactor vessel capsule 18 removed from the vessel in 1981 (Reference 7). This value is therefore not consistent with the ASTM E 185-82 definition for USE (average of points >95% shear). Further review of the 1984 Southwest Research Institute Report (Reference 7) and the 1975 Battelle report (Reference 6) discloses that the average of the two (2) unirradiated data points with >95% shear yields an upper shelf value of 124.5 ft-lbs, essentially the same as the original value of 125 ft-lbs.

An irradiated USE value of 90.5 ft-lbs obtained from capsule 18 was used in deriving the measured decrease in USE provided in Figure 4.2.1-8 of the LRA. This irradiated value was obtained by averaging two (2) Charpy impact data points for ESW with a shear value ≥95% found in Table V of the 1984 Southwest Research Institute report for Quad Cities Unit 2 (Reference 7). This value is consistent with the definition of upper shelf energy in ASTM E185-82. A value of 90 ft-lbs was previously reported for the irradiated USE in ComEd response to Generic Letter 92-01 (Reference 2). This value is slightly more conservative than the 90.5 ft-lbs used to calculate the %decrease reported in the LRA.

The EMA process defined in BWRVIP-74 was performed using the results from Capsule 18. This analysis is presented in the form of a revised LRA Table 4.2.1-8:

**Table 4.2.1-8: Equivalent Margin Analysis for Quad Cities Unit 2 Weld Material**

| **BWR/2-6 WELD** |
|---|
| Surveillance Weld USE:<br>%Cu = 0.14<br><br>1st Capsule Fluence = $1.69 \times 10^{16}$ n/cm2<br>2nd Capsule Fluence = $6.6 \times 10^{16}$ n/cm2<br><br>1st Capsule Measured % Decrease = 15 (Charpy Curves)<br>2nd Capsule Measured % Decrease = 28 (Charpy Curves)<br><br>1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2)<br>2nd Capsule R.G. 1.99 Predicted % Decrease = 9 (R.G. 1.99, Figure 2) |
| Limiting Beltline Weld USE:<br>%Cu = 0.24<br>54 EFPY 1/4T Fluence = $3.9 \times 10^{17}$ n/cm2<br>R.G. 1.99 Predicted % Decrease = 18.5(R.G. 1.99, Figure 2)<br>Adjusted % Decrease = 43 (R.G. 1.99, Position 2.2) |
| 43%> 39% , so vessel welds are not bounded by the BWRVIP-74 equivalent margin analysis |

Because the approach of BWRVIP-74 did not demonstrate equivalent margin, a plant-specific equivalent margin analysis was performed following the methodology of RG 1.161, Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb. The end of life upper shelf energy of 34.2 ft-lb for Quad Cities Unit 2 electroslag weld material was calculated based on a statistical lower bound unirradiated USE of 60 ft-lb (per BWRVIP-74), and the 43% drop calculated per R.G. 1.99, Position 2.2 from the limiting capsule results.

The analysis used inputs derived from approved GE Licensing Topical Report NEDO-32205-A, 10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels, in accordance with R.G. 1.161, ASME Section XI Appendix K, and ASME Code Case N-512-1. Normal, upset, emergency, and faulted loading conditions from the topical report were shown to be applicable for Quad Cities Unit 2.

The plant specific analysis showed that equivalent margin can be demonstrated for a USE of 32.43 ft-lbs. Since the limiting end of life USE for Quad Cities 2 vessel welds of 34.2 ft-lbs exceeds the required minimum value, this vessel meets the margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The plant specific equivalent margin analysis, GE-NE-0000-0027-0575-01, Rev. 0, The Upper Shelf Energy Evaluation for RPV Electroslag Welds at Quad Cities Unit 2, is provided as Attachment 3 of this submittal.

In addition to those references specifically discussed above, the references for all of the surveillance capsule results presented in the LRA are summarized in Table 4.2-1 which follows.

## References for USE values

**Table 4.2-1**

| | Dresden 2 | Dresden 3 | Quad Cities 1 | Quad Cities 2 |
|---|---|---|---|---|
| Unirradiated Properties | N/A for limiting ESW material | Dresden Nuclear Plant Unit No. 3 Vessel Surveillance Programs: Unirradiated Mechanical Properties, Battelle Columbus Laboratories, February 15, 1975 | Quad Cities Nuclear Plant Unit No. 1 and Unit No. 2 Reactor Pressure Vessel Surveillance Programs: Unirradiated Mechanical Properties, Battelle Columbus Laboratories, February 15, 1975 | Quad Cities Nuclear Plant Unit No. 1 and Unit No. 2 Reactor Pressure Vessel Surveillance Programs: Unirradiated Mechanical Properties, Battelle Columbus Laboratories, February 15, 1975 |
| 1st wall capsule results | Dresden Nuclear Power Station Unit One and Unit Two Mechanical Properties of Irradiated Reactor Vessel Material Surveillance Specimens, NEDC-12585, May, 1975 | Dresden Nuclear Plant Unit No. 3 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 13, Capsule Basket No. 14, and Neutron Dosimeter Monitor, Battelle Columbus Laboratories, March 1, 1975 | Quad Cities Nuclear Plant Unit No. 1 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 2, Capsule Basket No. 3, and Neutron Dosimeter Monitor, Battelle Columbus Laboratories, March 1, 1975 | Quad Cities Nuclear Plant Unit No. 2 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 12 and Capsule Basket No. 13, Battelle Columbus Laboratories, September 19, 1975 |
| 2nd wall capsule results | Dresden Nuclear Power Station Unit 2 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 8, Final Report SwRI Project No. 06-6901-002, March 1983 | Dresden Nuclear Plant Reactor Pressure Vessel Surveillance Program: Unit No. 3 Capsule Basket Assembly No. 6, Battelle Columbus Laboratories, June 15, 1979 | Quad Cities Nuclear Power Station Unit 1 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 8, Final Report SwRI Project No. 06-7857, August 1984 | Quad Cities Nuclear Power Station Unit 2 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 18, Final Report SwRI Project No. 06-7484-002, March 1984 |
| 3rd wall capsule results | N/A | Dresden Nuclear Power Station Unit 3 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 18, Final Report SwRI Project No. 06-7484-003, February 1984 | N/A | N/A |

2) Response to Question 2.

The peak fluence shown below is applied to the lower-intermediate shell and axial welds. For the lower shell, the peak fluence is adjusted by the axial flux distribution factor based on an elevation approximately 42 inches above the bottom of active fuel, which represents the lower to lower-intermediate girth weld. For the lower shell and welds and the lower to lower-intermediate girth weld, the axial factor of 0.71 is applied for pre-EPU, and the axial factor of 0.74 is applied for EPU conditions. The results for the lower shell, lower shell welds, and the lower to lower-intermediate girth weld are presented following the peak surface calculations.

Dresden 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (19.4/54) + 3.46e8 n/cm$^2$-s * 1.7e9 s * (34.6/54) = 5.67e17 n/cm$^2$
Dresden 2 54 EFPY Peak Lower Shell and Weld Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (19.4/54) * 0.71 + 3.46e8 n/cm$^2$-s * 1.7e9 s * (34.6/54) * 0.74 = 4.15e17 n/cm$^2$

Dresden 3 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (19.8/54) + 3.46e8 n/cm$^2$-s * 1.7e9 s * (34.2/54) = 5.67e17 n/cm$^2$
Dresden 3 54 EFPY Peak Lower Shell and Weld Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (19.8/54) * 0.71 + 3.46e8 n/cm$^2$-s * 1.7e9 s * (34.2/54) * 0.74= 4.14e17 n/cm$^2$

Quad Cities 1 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (21.1/54) + 3.46e8 n/cm$^2$-s * 1.7e9 s * (32.9/54) = 5.66e17 n/cm$^2$
Quad Cities 1 54 EFPY Peak Lower Shell and Weld Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (21.1/54) * 0.71 + 3.46e8 n/cm$^2$-s * 1.7e9 s * (32.9/54) * 0.74 = 4.13e17 n/cm$^2$

Quad Cities 2 54 EFPY Peak Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (21/54) + 3.46e8 n/cm$^2$-s * 1.7e9 s * (33/54) = 5.66e17 n/cm$^2$
Quad Cities 2 54 EFPY Peak Lower Shell and Weld Surface Fluence Calculation:
3.12e8 n/cm$^2$-s * 1.7e9 s * (21/54) * 0.71 + 3.46e8 n/cm$^2$-s * 1.7e9 s * (33/54) * 0.74 = 4.13e17 n/cm$^2$

With regard to the request "Describe how the pre-EPU axial flux profile compares with the post-EPU profile": This information was provided in GE Proprietary flux profile data that was submitted to the NRC in Memo, Exelon to NRC, "Additional Information Regarding Request for License Amendment for Pressure Temperature Limits, " dated July 31, 2003. Figure 2 of the July 31 memo provides the requested information.

3) Response to Question 3 and Open Item 4.2.1(c).

Tables 1 and 2 present the fluence and impact of fluence on the USE EMA % decrease as defined in Figure 2 of Regulatory Guide 1.99, Revision 2 (RG1.99) for the plate and weld materials, respectively. This is only intended to demonstrate the direct impact of fluence and does not include the RG1.99 Position 2.2 adjustment for cases where the measured % decrease exceeds the RG1.99 predicted % decrease. Position 2.2 is applied for the Dresden 2 plate material, Quad Cities 1 weld material, and the Quad Cities 2 weld material, as shown in Tables 3 and 4.

**Table 1 - Dresden and Quad Cities Plate Materials USE EMA**

| Plant | Plate Heat | Shell | 54 EFPY Surface Fluence ($x10^{17}$ $n/cm^2$) | 54 EFPY 1/4T Fluence ($x10^{17}$ $n/cm^2$) | Chemistry %Cu | 54 EFPY USE EMA (%decrease) | Bounding Material |
|---|---|---|---|---|---|---|---|
| Dresden 2 | A9128-2 | Lower | 4.2 | 2.9 | 0.20 | 14.5 | |
| | B3990-2 | Lower | 4.2 | 2.9 | 0.18 | 13.5 | |
| | A9128-1 | Lower | 4.2 | 2.9 | 0.20 | 14.5 | |
| | **B4065-1** | **Low-Int** | **5.7** | **3.9** | **0.23** | **15.5** | **Bounding** |
| | B5764-1 | Low-Int | 5.7 | 3.9 | 0.10 | 9 | |
| | B4030-1 | Low-Int | 5.7 | 3.9 | 0.20 | 14 | |
| | B4030-2 | Low-Int | 5.7 | 3.9 | 0.20 | 14 | |
| Dresden 3 | C1256-2 | Lower | 4.1 | 2.9 | 0.11 | 9 | |
| | B5159-2 | Lower | 4.1 | 2.9 | 0.24 | 15 | |
| | C1182-2 | Lower | 4.1. | 2.9 | 0.22 | 14 | |
| | **A0237-1** | **Low-Int** | **5.7** | **3.9** | **0.23** | **15.5** | **Bounding** |
| | B5118-1 | Low-Int | 5.7 | 3.9 | 0.22 | 15 | |
| | C1290-2 | Low-Int | 5.7 | 3.9 | 0.15 | 11.5 | |
| Quad Cities 1 | **B5524-1** | **Lower** | **4.1** | **2.9** | **0.27** | **16.5** | **Bounding** |
| | A0610-1 | Lower | 4.1 | 2.9 | 0.21 | 13.5 | |
| | C1485-2 | Lower | 4.1 | 2.9 | 0.23 | 14.5 | |
| | C1505-2 | Low-Int | 5.7 | 3.9 | 0.18 | 13 | |
| | C1498-2 | Low-Int | 5.7 | 3.9 | 0.17 | 12.5 | |
| | A0931-1 | Low-Int | 5.7 | 3.9 | 0.14 | 11 | |
| Quad Cities 2 | C1516-2 | Lower | 4.1 | 2.9 | 0.16 | 11.5 | |
| | **C1501-2** | **Lower** | **4.1** | **2.9** | **0.18** | **12** | **Bounding** |
| | C1722-2 | Lower | 4.1 | 2.9 | 0.14 | 10.5 | |
| | C2753-2 | Low-Int | 5.7 | 3.9 | 0.08 | 8 | |
| | C2868-1 | Low-Int | 5.7 | 3.9 | 0.08 | 8 | |
| | C3307-2 | Low-Int | 5.7 | 3.9 | 0.12 | 10 | |

Table 2 - Dresden and Quad Cities Weld Materials*

| Plant | Weld Heat or ID | Shell | 54 EFPY Surface Fluence ($x10^{17}$ n/cm$^2$) | 54 EFPY 1/4T Fluence ($x10^{17}$ n/cm$^2$) | Chemistry %Cu | 54 EFPY USE EMA (%decrease) | Bounding Material |
|---|---|---|---|---|---|---|---|
| Dresden 2 | **ESW** | **Low-Int** | **5.7** | **3.9** | **0.24** | **18.5** | **Bounding** |
| | 1P0661/8304 | Low-Int | 5.7 | 3.9 | 0.17 | 15 | |
| | 1P0815/8350 | Low-Int | 5.7 | 3.9 | 0.17 | 15 | |
| | ESW | Lower | 4.2 | 2.9 | 0.24 | 17.5 | |
| | 1P0815/8304 | Lower | 4.2 | 2.9 | 0.17 | 14 | |
| | 71249/8504 | Lower to Low-Int (Girth) | 4.2 | 2.9 | 0.23 | 17 | |
| Dresden 3 | ESW | Low-Int | 5.7 | 3.9 | 0.24 | 18.5 | |
| | ESW | Lower | 4.1 | 2.9 | 0.24 | 17.5 | |
| | **299L44/8650** | **Lower to Low-Int (Girth)** | **4.1** | **2.9** | **0.34** | **21.5** | **Bounding** |
| Quad Cities 1 | ESW | Low-Int | 5.7 | 3.9 | 0.24 | 18.5 | |
| | ESW | Lower | 4.1 | 2.9 | 0.24 | 17.5 | |
| | 72445/8688 | Lower to Low-Int (Girth) | 4.1 | 2.9 | 0.22 | 16.5 | |
| | **406L44/8688** | **Lower to Low-Int (Girth)** | **4.1** | **2.9** | **0.27** | **18.5** | **Bounding**** |
| Quad Cities 2 | **ESW** | **Low-Int** | **5.7** | **3.9** | **0.24** | **18.5** | **Bounding** |
| | ESW | Lower | 4.1 | 2.9 | 0.24 | 17.5 | |
| | S3986/3870 Linde 124 | Lower to Low-Int (Girth) | 4.1 | 2.9 | 0.05 | 8.5 | |

* %decrease values from Figure 2 of RG1.99, rounded to the next highest 0.5%

** The decrease value prior to rounding up for the ESW Low-Int weld material is 18.1%; for Heat 406L44/8688 is 18.3%.

**Table 3 - Dresden 2 & Quad Cities 1 USE EMA Adjusted**

| Plant | Weld or Plate | Measured % Decrease | RG1.99 Predicted % Decrease | 54 EFPY % Decrease from Figure 2 of RG1.99 | Adjusted % Decrease Using Position 2.2 of RG1.99 |
|---|---|---|---|---|---|
| Dresden 2 | Plate | 8 | 6 | 15.5 | 17.5 |
| Quad Cities 1 | Weld | 12 | 10 | 18.5 | 17.5* |

* Because the adjusted % decrease is less than that without the Position 2.2 adjustment, the higher value is considered for qualification.

**Table 4 - Quad Cities 2 USE Adjusted***

| Plant | Weld or Plate | Measured % Decrease | RG1.99 Predicted % Decrease | 54 EFPY % Decrease from Figure 2 of RG1.99 | Adjusted % Decrease from Position 2.2 of RG1.99 |
|---|---|---|---|---|---|
| Quad Cities 2 | Weld (capsule 2) | 28 | 9 | 18.5 | 43 |

As discussed in the response to question 1, a plant-specific equivalent margin analysis was performed for Quad Cities 2 to demonstrate equivalent margin for the limiting end of life USE.

Supporting References:

1) Southwest Research Institute, Dresden Nuclear Power Station Unit 2 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 8, Final Report SwRI Project No. 06-6901-002, March 1983.
2) Letter, Commonwealth Edison (Marcia A. Jackson) to NRC (NRR), "Dresden Station Units 2&3, Quad Cities Station Units 1&2, LaSalle County Station Units 1 and 2, NRC Docket Nos. 50-237/249, 50-254/265, & 50-373/374," Attachment B "CECo Response to Generic Letter 92-01, Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2," dated July 1, 1992.
3) Southwest Research Institute, Dresden Nuclear Power Station Unit 3 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 18, Final Report SwRI Project No. 06-7484-003, February 1984.
4) Final Report on Dresden Nuclear Plant Unit No. 3 Vessel Surveillance Programs: Unirradiated Mechanical Properties, Battelle Columbus Laboratories, February 15, 1975.
5) Southwest Research Institute, Quad Cities Nuclear Power Station Unit 1 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 8, Final Report SwRI Project No. 06-7857, August 1984.
6) Final Report on Quad Cities Nuclear Plant Unit No. 1 and Unit No. 2 Reactor Pressure Vessel Surveillance Programs: Unirradiated Mechanical Properties, Battelle Columbus Laboratories, February 15, 1975.

7) Southwest Research Institute, Quad Cities Nuclear Power Station Unit 2 Reactor Vessel Irradiation Surveillance Program Analysis of Capsule No. 18, Final Report SwRI Project No. 06-7484-002, March 1984.
8) Final Report on Quad Cities Nuclear Plant Unit No. 2 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 12 and Capsule Basket No. 13, Battelle Columbus Laboratories, September 19, 1975.
9) Dresden Nuclear Power Station Mechanical Properties of Unirradiated Reactor Vessel Material Surveillance Specimens, GE Report NEDC-12575, April 1975.
10) Dresden Nuclear Power Station Unit One and Unit Two Mechanical Properties of Irradiated Reactor Vessel Material Specimens, GE Report NEDC-12585, May 1975.
11) GE-NE-0000-0027-0575-01, Rev. 0, The Upper Shelf Energy Evaluation for RPV Electroslag Welds at Quad Cities Unit 2, March 2004.

Attachment 2

Comments on the SER text

**Draft SER Comments Section 1**

1. The resolution and documentation necessary to close the open items and confirmatory items is contained in Attachment 1 of this submittal.

**Draft SER Comments Section 2**

1. In section 2.1 of the draft SER, Part b of the subsection titled Application of the Scoping Criteria in 10 CFR 54.4.(a)(2), (page 2-21) states that the NRC requires additional clarification regarding the capability of active and passive safety-related SSCs located greater than 20 feet from a potential spray source to tolerated wetting. This section of the draft SER also refers to Open Item 2.1-1. The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to the open item.

2. In section 2.1 of the draft SER, Part e of the subsection titled Application of the Scoping Criteria in 10 CFR 54.4.(a)(2), (page 2-23) states that the NRC requested justification that would demonstrate that failure of the non-safety related piping that was potentially excluded from the scope of license renewal would not adversely impact the safety-related portion of the piping system in accordance with 10 CFR 54.4.(a)(2). This section of the draft SER also refers to Open Item 50-237/03-04-01. The resolution and documentation necessary to close this open item was contained in a letter to the NRC dated March 25, 2004 and is also contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to the open item.

3. The last paragraph in section 2.1 of the draft SER, in the subsection titled Application of the Scoping Criteria in 10 CFR 54.4.(a)(2), (page 2-24) refers to Open Item 2.1-1 and Open Item 50-237/03-04-01. This section of the SER should be updated to address the resolution of these open items.

4. In section 2.4.1.2 of the SER, the response to RAI 2.4-2 discussed on pages 2-227 and 2-228 does not incorporate the corrections noted in the Supplemental RAI response. The Supplemental RAI response submitted by letter dated December 5, 2003, should be incorporated into the SER after the response to RAI 2.4-2 on page 2-228.

5. In Section 2.4.1.2 (page 2-230), the SER states that 'The staff finds the applicant's response acceptable and, therefore, RAI 2.4-3 is resolved.' Note however that the response to Supplemental RAI 2.4-3 is not included. This section of the SER should be updated to include the Supplemental SER response, submitted by letter dated December 12, 2003, following the RAI 2.4-3 response on page 2-230.

6. In section 2.4.2.2 of the SER, paragraph (b) on page 2-234, the negative pressure value listed in the second sentence should read "≥ 1/4 inch H2O."

7. In section 2.4.3.2 of the SER, the following new table entry for caulking/sealants is included in RAI 2.3.2.9-3 but is missing on page 2-237.

| Component | Component Intended Function | Aging Management Ref No. |
|---|---|---|
| Caulking/Sealants | Structural Pressure Barrier | 3.5.2.3 |

8. In section 2.4.11.2, discussion of RAI 2.4-7 response (page 2-257) - The boundary diagram reference numbers contain typos under "Quad Cities (a)" and should read as follows: "LR-QDC-M-37" and "LR-QDC-M-79."

9. Same section, discussion of supplemental RAI 2.4-7 response (page 2-262) – Clarifications provided by the Supplemental RAI response submitted by letter dated December 5, 2003 should be added following the table on page 2-262.

10. The table provided in SER Section 2.4.16.2 (pages 2-279 and 2-280) as response to RAI 2.4-11 identifies those systems and locations of insulation and/or insulation jacketing within the scope of license renewal. The SER table is missing the following row from the RAI response table:

| System | Location | System | Location |
|---|---|---|---|
| Reactor core isolation cooling system (Quad Cities only) | Inside and Outside Containment | HVAC – radwaste building | Outside Containment |

11. SER Section 2.5.1.2 (page 2-283) does not address RAI 2.5-1. The SER should add a discussion of the RAI response submitted by letter dated October 3, 2003.

**Draft SER Comments Section 3**

1. In SER section 3.0.3, there are typos in the section numbers for Sections 3.0.3.2.2 and 3.0.3.2.3. On page 3-10, the Section number appears as 3.0.3.1.2 but should be 3.0.3.2.2. Similarly, on page 3-13 the Section number appears as 3.0.3.1.3 but should be 3.0.3.2.3.

2. The response to RAI B.1.2d and Supplemental RAI B.1.2d in Section 3.0.3.2.2, the second-to-last paragraph of page 3-12 and Section 3.0.3.2.3, page 3-13 refers to Confirmatory Item B.1.2-1. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

3. RAI B.1.13(g) and Supplemental RAI B.1.13(a) address inspections for managing galvanic corrosion. Section 3.0.3.6.2 (page 3-26) does not incorporate the corrections noted in the Supplemental RAI B.1.13(a) response submitted by letter dated December 17, 2003. For instance, the original response stated: "The aging effects of galvanic corrosion are managed through periodic heat exchanger inspections."

The above statement was corrected in the supplemental response to read as follows: "The aging effects of galvanic corrosion are managed through periodic inspections of in-scope components as appropriate."

4. For the Supplemental RAI response dated December 12, 2003 in Section 3.0.3.9.2 (page 3-35), the RAI response letter dated December 12, 2003, does not contain information associated with the aluminum tanks at Dresden. The wording as summarized in the SER is also unclear and confusing. If the NRC intended to include the response dated December 22, 2003 (Supplemental RAI B.1.2), then the SER text should be as follows:

   > "There was no definitive aging mechanism identified for degradation of the subject Dresden aluminum tank bottoms. The Above Ground Carbon Steel Tanks Program 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."

5. RAI B.1.23-2 and Supplemental RAI B.1.23, and B.1.23-2.1 thru B.1.23-2.6 responses are summarized in Section 3.0.3.10.2, (pages 3-37 & 3-38). SER RAI response item (5) at the bottom of page 3-38 refers to Lubricating Oil Monitoring Activities "...expanded to include the analysis of the turbine oil systems components (Dresden only) exposed to generator hydrogen seal oil ...." Per the supplemental RAIs, the turbine oil systems for both stations are in scope and will be managed by the Lubricating Oil Monitoring System with the One-Time Inspection Program used to verify its effectiveness. The words '(Dresden only)' should be deleted from SER RAI response item (5).

6. Section 3.0.3.10.2 refers to Confirmatory Item B.1.23-1. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

7. Also in Section 3.0.3.10.2, (page 3-39), Part b of RAI B.1.23-2 refers to Open Item B.1.23-2. The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the open item.

8. RAI B.1.25 and Supplemental RAI B.1.25 responses are addressed in Section 3.0.3.12.2, (pages 3-44 & 3-45). This section refers to Confirmatory Item B.1.25-1. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

9. Section 3.0.3.14, Supplemental RAI B.1.30 includes the response to Confirmatory Item 3.0.3.14.2-1. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

10. Section 3.0.3.16 refers to Confirmatory Item B.2.5-1. Supplemental RAIs B.1.23.2.3 and B.1.23.2.4 provide the requested information. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

11. In section on page 3-86, the SER states, "At the completion of the review noted in item 1 above, Exelon will provide a list of commitments to the applicable BWRVIP documents or identify exceptions taken. This is part of Commitment #3 in Appendix A of this SER." This reference to Commitment #3 is not correct. The suggested correction is:

    "At the completion of the review noted in item 1 above, Exelon will provide a list of commitments to the applicable BWRVIP documents or identify exceptions taken. This is part of Commitment #6 in Appendix A of this SER."

12. Section 3.1.2.3.2 on page 3-87 refers to Confirmatory Item 3.1.2.3.2-1. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to this confirmatory item. Additional references to the same confirmatory item contained on pages 3-88, 3-92, 3-93, 3-94, 3-98, 3-114, and 3-118 should also be updated.

13. The first paragraph of section 3.1.2.3.6 on page 3-96 states that "The staff finds this response acceptable because the rim hold-down bolts no longer provide structural load and do meet the definition of a TLAA as defined in 10 CFR 54.3(a)(3) and (5)." The stress relaxation of the rim hold-down bolts is not a TLAA for Dresden and Quad Cities. The suggested correction is:

    "The staff finds this response acceptable because the rim hold-down bolts no longer provide structural load and do not meet the definition of a TLAA as defined in 10 CFR 54.3(a)(3) and (5)."

    The same paragraph refers to Confirmatory Item 3.1.2.3.6-1 on the same topic. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to this confirmatory item.

14. In section 3.1.2.3.8 on page 3-100, the last paragraph states: "the Exelon Reactor Vessel Surveillance Program will be enhanced to incorporate the reactor vessel surveillance program consistent with the staff-approved versions of BWRVIP-78 and BWRVIP-86. The applicant further stated that this commitment is already included in Section A.1.22 of the Dresden and Quad Cities UFSAR Supplement." As part of the 2004 annual update, Exelon revised this section of the UFSAR Supplement to reference BWRVIP-86A for 32 EFPY and BWRVIP-116 for the period of extended

operation. The approved version of BWRVIP-86 incorporates the relevant program elements of BWRVIP-78, therefore BWRVIP-78 was removed from the Supplement.

15. In section 3.1.2.3.8 on page 3-100, the last paragraph states: "On the basis of its review and audit of the applicant's program, pending satisfactory resolution of Confirmatory Item 3.1.2.3.8-1..." Earlier in the same section, the NRC refers to the Exelon submittal of November 21, 2003, committing to provide a plant-specific surveillance plan if the NRC does not approve BWRVIP-116. This commitment is captured as Commitment #22 in Appendix A of the SER. Since the commitment is explicit and provides satisfactory resolution, there is no apparent reason to keep this Confirmatory Item open. Exelon recommends that it be removed from the SER.

16. In section 3.2.1.4.1, page 3-107 and 108, the NRC describes Confirmatory Item 2.3.4.2-3 concerning the CRD return line nozzles. Exelon responded to this item by letter dated January 26, 2004. The SER should be updated to reflect this response. The same Confirmatory Item is discussed in the second paragraph on page 112, which should also be updated.

17. In section 3.2.1.4.1, page 3-111, third paragraph, the SER states in part, "However, the applicant's identification of no aging effect for the external surface of carbon steel components exposed to a containment nitrogen environment is not acceptable because the BWR containment environment typically has high humidity." The following paragraph states that "The staff finds the applicant's identification of no loss of material for the carbon steel components exposed to a containment environment acceptable...." These paragraphs are in conflict as written. Exelon suggests that the third paragraph be revised to read as follows:

    > "However, the staff questioned the applicant's identification of no aging effect for the external surface of carbon steel components exposed to a containment nitrogen environment because the BWR containment environment typically has high humidity."

    A similar comment applies to the discussion of the same issue in the last paragraph on page 3-132, where the SER states "This identification of no aging effect is not acceptable..." and later, "The staff agrees with the applicant that there are no applicable aging effects..."

18. In section 3.2.1.4.1, page 3-116, the SER states: "According to Table 2-1 of BWRVIP-76, when fluences exceed $5x10^{20}$ n/cm$^2$, a plant-specific analysis is required to be submitted to the NRC. The applicant needs to submit this analysis to the staff (Confirmatory Item 3.1.2.4.2-1)."

    The response to this Confirmatory Item is included in Attachment 1 of this submittal. Exelon recommends this Confirmatory Item be updated accordingly (including a later reference on page 3-118).

19. In section 3.2.1.4.2, page 3-125, the SER states, "As noted in the response to RAI 3.1-24b, the IHSI treatment of the susceptible welds was not effective in mitigating IGSCC. The applicant also stated that no information is yet available on the effectiveness of noble metal chemical injections on IGSCC mitigation, but the use of

HWC appears to provide a beneficial effect." This RAI reference is not correct for these statements. The suggested correction is:

> "As noted in the response to RAI B.1-7a, the IHSI treatment of the susceptible welds was not effective in mitigating IGSCC. The applicant also stated that no information is yet available on the effectiveness of noble metal chemical injections on IGSCC mitigation, but the use of HWC appears to provide a beneficial effect."

20. In section 3.2.2.4.6, page 3-187, the SER states, "The staff's discussion of this RAI and its resolution by the applicant are provided in section 3.2.2.6 of this SER." The SER does not include a section 3.2.2.6, and the correct reference could not be readily identified.

21. The NRC Evaluation part of Section 3.3.2.3.2 on page 3-215 refers to Confirmatory Item B.1.17. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

22. The NRC Evaluation part of Section 3.3.2.3.7 on page 3-231 refers to Confirmatory Item B.1.23-2.5. The resolution and documentation necessary to close this confirmatory item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in response to the confirmatory item.

23. Section 3.3.2.4.11 (pages 3-274 & 3-275) of the SER contains a summary of the RAI 3.3.2.4.11(a) and Supplemental RAI 3.3.2.4.11 responses, however the date of the Supplemental RAI response is missing. The fourth sentence of the second paragraph on page 3-274 should be revised to read "In its response dated October 3, 2003 and Supplemental RAI 3.3.2.4.11 response dated December 22, 2003, the applicant......"

24. In Section 3.3.2.4.21, (pages 3-306 & 3-307), the statement at the end of the second-to-last paragraph on page 3-307 contains a typo. It currently states that the NRC finds the applicant's response to RAI 3.3.2.4.16 acceptable. The RAI number should read "RAI 3.3.2.4.21."

25. In section 3.4.2.4.1 on page 3-350, the SER states: "Therefore, in RAI 3.2.4.1-3, the staff asked the applicant to clarify the environment with respect to temperature, radiation levels, and time when the containment is not or has not been inerted, to justify that neoprene hoses do not require aging management." This RAI reference is incorrect. The suggested correction is:

> "Therefore, in RAI 3.4.1-3, the staff asked the applicant to clarify the environment with respect to temperature, radiation levels, and time when the containment is not or has not been inerted, to justify that neoprene hoses do not require aging management."

26. Section 3.4.2.4.1 on page 3-351 refers to Open Item B.1.23-2. The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information

contained in the response to this open item. Additional references to the same open item contained on page 3-359 and 3-361 should also be updated.

27. Section 3.4.2.4.5 on page 3-359 refers to confirmatory item B.1.23-1. The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to this confirmatory item. Additional references to the same confirmatory item contained on page 3-360 should also be updated.

28. SER Section 3.5.2.2.2 (page 3-384) contains typos. "See Open Item 3.5.2.3.2-1" should read "Open Item 3.5.2.3.2.2-1" and "in SER Section 3.5.2.3.2" should read "SER Section 3.5.2.2.3".

29. The Supplemental RAI 3.5-7 response lists two Open Items, 3.5.2.2.2.1-1 and 3.5.2.3.2.2-1. Open Item 3.5.2.2.2.1-1 is not noted in the draft SER whereas Open Item 3.5.2.3.2.2-1 is referenced to SER Section 3.5.2.2.3. Section 3.5.2.2.2 (page 3-284) of the SER should be updated to include the information contained in the Supplemental RAI 3.5-7 response to Open Item 3.5.2.2.2.1-1. Section 3.5.2.2.3 (page3-387) should be updated to include the information in response to Open Item 3.5.2.3.2.2-1. The resolution and documentation necessary to close open item 3.5.2.3.2.2.1-1 is contained in Attachment 1 of this submittal.

30. The SER on page 3-393 refers to Open Item 3.5.2.3.2-1. The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. Section 3.5.2.3.3 of the SER should be updated to include the information contained in response to the open item.

31. The response to RAI 3.5-10 was accurately incorporated in Section 3.5.2.4.1, (pages 3-407 & 3-408). The Supplemental RAI 3.5-10 response was not included in the draft SER. A discussion of this response submitted by letter dated December 12, 2003 should be added following the response to RAI 3.5-10.

32. Supplemental RAI 3.5-16 was not addressed in Section 3.5.2.4.5 (page 3-427). A discussion of this response submitted by letter dated December 5, 2003 should be added following the response to RAI 3.5-16.

**Draft SER Comments Section 4**

1. Section 4.2.1.1 of the draft SER correctly states "The applicant stated that a report summarizing the results of the equivalent margin analysis will be submitted for NRC approval by December 31, 2003." As a result of continuing interaction with the NRC Exelon provided updated results of the equivalent margin analysis, including a new plant-specific EMA for Quad Cities 2, in the response to RAI 4.2.1 which is summarized in Attachment 1 of this submittal.

2. Section 4.2.1.6 of the draft SER states that the applicant is required to submit an update to LRA Section 4.2.6 to include Quad Cities vessel circumferential weld examination relief analysis is accordance with 10 CFR 54.3(a) upon NRC's approval of the May 16, 2003, relief request. This section of the draft SER also refers to Confirmatory Item 4.2.1.6. The resolution and documentation necessary to close this

confirmatory item was provided in the annual update required by 10 CFR 54.21(b) sent to the NRC in a letter dated March 5, 2004. This documentation is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to the confirmatory item and the information contained in the letter dated March 5, 2004.

3. Section 4.2.1.7 of the draft SER should be updated to include information concerning Quad Cities reactor vessel axial weld failure probability provided to the NRC in the annual update required by 10 CFR 54.21(B) sent to the NRC in a letter dated March 5, 2004. This documentation is contained in Attachment 1 of this submittal.

4. Section 4.2.2.1 of the draft SER (page 4-9) discusses Confirmatory Item 4.2.1(a). The resolution and documentation necessary to close this open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to the confirmatory item.

5. Section 4.2.2.1 of the draft SER refers to Confirmatory Item 4.2.1 (page 4-10) and open item 4.2.1(c) (page 4-11). The resolution and documentation necessary to close this confirmatory item and open item is contained in Attachment 1 of this submittal. This section of the SER should be updated to include the information contained in the response to both the open item and the confirmatory item.

6. On page 4-10, the discussion of equivalent margin for limiting beltline welds and plates at one point refers to "Quad Cities Unit 3." The correct designation is Quad Cities Unit 2. This comment also applies to Confirmatory Item 4.2.1 on page 1-11 of the SER.

7. Section 4.2.2.6 of the draft SER (page 4-16) should be updated to include information concerning Quad Cities reactor vessel axial weld failure probability provided to the NRC in the annual update required by 10 CFR 54.21(B) sent to the NRC in a letter dated March 5, 2004. This documentation is contained in Attachment 1 of this submittal.

8. Section 4.2.2.7 of the draft SER (page 4-18) should be updated to include information concerning Quad Cities reactor vessel axial weld failure probability provided to the NRC in the annual update required by 10 CFR 54.21(B) sent to the NRC in a letter dated March 5, 2004. This documentation is contained in Attachment 1 of this submittal.

**Draft SER Comments Appendix A**

1. Commitment no. 6: the review of applicable BWRVIPs is documented in the response to Confirmatory Item 3.1.2.3.2-1, provided in Attachment 1 of this submittal.

2. Commitment no. 12: the reference for the response to RAI B.1.12 is letter RS-03-181, dated October 3, 2003.

3. Commitment no. 25: the reference for the response to supplemental RAI B.1.25-1 is letter RS-04-046 dated March 25, 2004.

4. Commitment no. 26: the reference for the response to supplemental RAI B.1.27 is letter RS-04-046 dated March 25, 2004.

5. Commitment no. 27: the reference for the response to supplemental RAI B.1.27 is letter RS-04-046 dated March 25, 2004.

6. Commitment no. 30: the reference for the response to supplemental RAI B.1.27 is letter RS-04-046 dated March 25, 2004.

7. Commitment no. 47: the source information for this item should be copied from the Exelon letter RS-04-020 dated February 3, 2004.

**Draft SER Comments Appendix B**

Add the following letters:

- October 15, 2003 Letter (RS-03-201) from Mr. Patrick R. Simpson, Exelon, to the NRC, submitting the requested additional information (Accession No. ML033010396)

- February 3, 2004 Letter (RS-04-020) from Mr. Patrick R. Simpson, Exelon, to the NRC, Consolidated List of Commitments for License Renewal (Accession No. ML040420164)

- March 5, 2004 Letter (RS-04-039) from Mr. Patrick R. Simpson, Exelon, to the NRC, Amendment to the Application for Renewed Operating Licenses for Dresden and Quad Cities Nuclear Power Stations (Accession No. MLxxxxx)

- March 25, 2004 Letter (RS-04-46) from Mr. Patrick R. Simpson, Exelon, to the NRC, submitting supplemental information (Accession No. MLxxxxx)

- April 9, 2004 Letter (RS-04-057) from Mr. Patrick R. Simpson, Exelon, to the NRC, submitting responses to the draft SER Open Items and Confirmatory Items (Accession No. MLxxxxx)

Attachment 3

General Electric Upper Shelf Energy Evaluation Report