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10 CFR 50
10 CFR 51
10 CFR 54

RS-14-176

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Responses to NRC Requests for Additional Information, Set 29, dated June 4, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application

References:
1. Letter from Michael P. Gallagher, Exelon Generation Company LLC (Exelon) to NRC Document Control Desk, dated May 29, 2013, "Application for Renewed Operating Licenses"

2. Letter from Lindsay R. Robinson, US NRC to Michael P. Gallagher, Exelon, dated June 4, 2014, "Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 29 (TAC NOS. MF1879, MF1880, MF1881, and MF1882)"

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted the License Renewal Application (LRA) for the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (BBS). In Reference 2, the NRC requested additional information to support staff review of the LRA.

Enclosure A contains the responses to these requests for additional information.

Enclosure B contains updates to sections of the LRA affected by the response.

There are no new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. Al Fulvio, Manager, Exelon License Renewal, at 610-765-5936.

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

Executed on 06-18-2014

Respectfully,

Michael P. Gallagher
Vice President - License Renewal Projects
Exelon Generation Company, LLC

Enclosures: A. Responses to Requests for Additional Information
B. Updates to affected LRA sections

cc: Regional Administrator – NRC Region III
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
NRC Project Manager, NRR-DORL-Braidwood and Byron Stations
Illinois Emergency Management Agency - Division of Nuclear Safety

Enclosure A

Byron and Braidwood Stations (BBS), Units 1 and 2
License Renewal Application

Responses to Requests for Additional Information

RAI B.2.1.5-1a
RAI B.2.1.5-2a

RAI B.2.1.5-1a

Applicability:

Byron Station (Byron) and Braidwood Station (Braidwood), all units

Background:

By letter dated January 13, 2014, the applicant responded to request for additional information (RAI) B.2.1.5-1, which addressed loss of material due to wear of control rod drive mechanism (CRDM) nozzles resulting from interactions with CRDM nozzle thermal sleeves. The applicant stated that it is planning to demonstrate, using analysis, that the CRDM nozzle wear will not exceed a minimum safe value such that examinations will not be required during the period of extended operation (PEO). The applicant further stated that when completed, the analyses will include a detailed ASME Code evaluation of the CRDM housing with a reduced wall thickness using the bounding CRDM loads and transients. In addition, all ASME Code stress categories will be evaluated utilizing a finite element analysis and will explicitly consider all conditions to which the CRDM housing is subjected during normal and upset conditions.

Issue:

The staff cannot determine the adequacy of the applicant's analysis since this analysis has yet to be completed. Additional information is necessary to confirm that the applicant's analysis has an adequate technical basis and that analytical results are acceptable for managing CRDM nozzle wear.

Request:

1. Describe the technical basis of the analysis and specific references for the acceptance criteria of the analysis (e.g., ASME Code Section III Edition and paragraphs and current license basis document sections). As part of the response, confirm whether the acceptance criteria adequately addresses the design, normal, upset, emergency, fault, testing, and cyclic (i.e., fatigue analysis) conditions in updated final safety analysis report (UFSAR) Section 3.9 and its subsections.

2. Upon completion of the CRDM nozzle wear analysis, provide the results confirming that the wear indications meet the acceptance criteria discussed in Request Part 1.

   If the analysis finds that the acceptance criteria cannot be met for the maximum possible wear depth of 0.1075 inches, clarify whether volumetric examinations will be performed to monitor the wear depths for adequate aging management.

3. Provide any necessary updates to the license renewal application (LRA), consistent with the applicant's response to Request Parts 1 and 2 (e.g., revisions to the time-limited aging analyses and UFSAR supplements in the LRA).

Exelon Response:

1. Exelon is participating in the Westinghouse Owners' Group project, which is expected to provide a detailed analysis justifying that the nozzle wear acceptance criteria can be met for the maximum possible wear depth of 0.1075 inches. Based on the completed feasibility study for this project, preliminary evaluations of the stresses and fatigue usages were performed to determine the approximate wear depth that could be qualified according to ASME Code, Section III, Subsection NB. It is expected that the detailed analysis will be successful in qualifying the maximum possible nozzle wear depth of 0.1075 inches. The detailed analysis is scheduled to be completed in October 2014, and the results are expected to be communicated to the NRC by the end of November 2014. As part of the detailed analysis, the acceptance criteria is expected to adequately address the design, normal, upset, emergency, fault, testing, and cyclic (i.e., fatigue analysis) conditions in the updated final safety analysis report (UFSAR) Section 3.9 and its subsections.

2. The results from CRDM nozzle wear analysis are expected to confirm that the wear indications meet the acceptance criteria discussed in response to Request 1. The detailed analysis results are expected to be communicated to the NRC by the end of November 2014. If the detailed analysis finds that the acceptance criteria cannot be met for the maximum possible wear depth of 0.1075 inches, Exelon will work with the industry to develop an approved method of volumetrically examining the wear area of the CRDM housing.

3. It is expected that there will be no changes to the license renewal application (LRA) resulting from this analysis. This will be confirmed upon completion of the CRDM nozzle wear analysis described above.

RAI B.2.1.5-2a

Applicability:

Byron and Braidwood

Background:

By letter dated January 13, 2014, the applicant responded to RAI B.2.1.5-2, which addressed loss of material due to wear of CRDM nozzle thermal sleeves. The applicant indicated that based on the current examination results at Byron and Braidwood, none of the evaluated thermal sleeve indications approach the minimum wall thickness. The applicant also stated that as a result of the initial visual examinations at each unit, the five thermal sleeves with the worst wear were selected to be examined with ultrasonic testing (UT) in order to obtain measurements of the wear indications. In addition, the applicant indicated the applicant's ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD, program will monitor the depths of these worst wear indications for aging management.

Issue:

It is not clear to staff that the initial examinations detected the worst wear locations because the applicant's response did not specifically state where on the thermal sleeves the ~~worst~~ wear was located. In addition, the applicant's response does not include revisions to the UFSAR supplement (LRA Section A.2.1.1) for the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD, program to specify the inspections of the thermal sleeves, as described in the applicant's response to RAI B.2.1.5-2.

Request:

1. Describe the locations of the thermal sleeve wear to confirm that the initial visual examinations were capable of detecting the worst wear indications.

2. Justify why the applicant's response does not include revisions to the UFSAR supplement (LRA Section A.2.1.1) to identify the additional inspections of the thermal sleeves. Alternatively, revise the UFSAR supplement to identify the additional inspections of the thermal sleeves.

Exelon Response:

1. The wear indications on the thermal sleeves are located in the area where the thermal sleeve exits the control rod drive mechanism (CRDM) head adapter tube (i.e., CRDM housing). This location is made visible when the reactor vessel head is removed. Therefore, these visual examinations were capable of detecting the worst wear indications. The wear on the thermal sleeves at this location is attributed to the thermal sleeve contacting the inside diameter of the CRDM head adapter tube due to a flow-induced impact/rotational motion of the thermal sleeve. These wear indications were discovered while the J-groove weld examinations were being conducted under the reactor vessel head. As a result of similar findings at other PWR units, Westinghouse issued technical bulletin, TB-07-2, “Reactor Vessel Head Adapter Thermal Sleeve Wear”, to inspect the thermal

sleeve wear on the outer two (2) concentric rows of the CRDM housings. Subsequently, all the Byron and Braidwood units conducted visual examinations on all the thermal sleeves, and determined which five (5) at each unit had the worst wear. These five (5) designated thermal sleeves had ultrasonic testing (UT) performed to measure the wear depth on Byron Units 1 and 2, and Braidwood Unit 2. The first ultrasonic testing on Braidwood Unit 1 will be performed in the Spring 2015 Refueling Outage. For examinations performed at three (3) units so far, the reactor core positions where the worst wear occurred are within the outermost two (2) concentric rows, as indicated in the technical bulletin and WCAP-16911-P, “Reactor Vessel Head Thermal Sleeve Wear Evaluation for Westinghouse Domestic Plants.”

2. The current program for inspection of the CRDM thermal sleeves was outlined in the previous RAI B.2.1.5-2 in Exelon Letter RS-14-002, dated January 13, 2014. LRA Section A.2.1.1 and Section B.2.1.1 are revised, as shown in Enclosure B, to reflect the implementation of this plan.

Enclosure B

Byron and Braidwood Stations, Units 1 and 2

License Renewal Application Updates to affected LRA Sections

Resulting from the responses to the following RAI:

RAI B.2.1.5-2a

Note: To facilitate understanding, portions of the original LRA have been repeated in this Enclosure, with revisions indicated. Existing LRA text is shown in normal font. Changes are highlighted with ***bolded italics*** for inserted text.

As a result of changes to the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program identified in the response to B.2.1.5-2a, LRA Appendix A, Section A.2.1.1, page A-10 is revised as shown below. Revisions are indicated with ***bolded italics*** for inserted text:

A.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is an existing condition-monitoring program that consists of periodic volumetric, surface, and/or visual examinations of ASME Class 1, 2, and 3 pressure-retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting for assessment, identification of signs of age-related degradation, and establishment of corrective actions. The program includes examinations and tests performed to identify and manage cracking, loss of fracture toughness, and loss of material in Class 1, 2, and 3 piping and components. This ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is implemented in accordance with 10 CFR 50.55a and ASME Code, Section XI. These activities include examinations, testing, detection, monitoring and trending, and evaluation of results to confirm that aging effects are managed during the period of extended operation.

The control rod drive mechanism (CRDM) thermal sleeves are examined under an augmented ISI inspection program. The scope of examination is to ultrasonically test (UT) the five (5) thermal sleeves with the worst wear on each unit. The plan for managing thermal sleeve wear is to obtain measured (UT) wear data points on each unit at the five (5) designated thermal sleeve reactor core locations during three (3) different outages. The frequency for inspection of the reactor vessel head thermal sleeve for loss of material due to wear will be re-evaluated after the accumulation of the three (3) data points on each of the five (5) designated thermal sleeves. The three (3) series of examinations will be performed prior to the period of extended operation. Subsequently, the required frequency for further inspections, if required, will be determined using the guidance provided in WCAP-16911-P, "Reactor Vessel Head Thermal Sleeve Wear Evaluation for Westinghouse Domestic Plants."

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program will be enhanced to:

1. Conduct a visual inspection of the accessible portions of the ASME Class 2 reactor vessel flange leakage monitoring tube every other refueling outage.

This enhancement will be implemented prior to the period of extended operation.

As a result of changes to the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program identified in the response to B.2.1.5-2a, LRA Appendix B, Section B.2.1.1, page B-14 is revised as shown below. Revisions are indicated with ***bolded italics*** for inserted text:

B.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

Program Description

The existing ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program manages the aging effects of cracking, loss of fracture toughness, and loss of material in Class 1, 2, and 3 piping and components exposed to air with borated water leakage, reactor coolant, reactor coolant and neutron flux, treated borated water, steam, and treated water environments. This condition monitoring program includes periodic visual, surface, and volumetric examination and leakage testing of Class 1, 2, and 3 pressure-retaining components including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting for assessment, identification of signs of age-related degradation, and establishment of corrective actions. The program includes examinations and tests performed to identify and manage cracking, loss of fracture toughness, and loss of material in Class 1, 2, and 3 piping and components. Inspection of these components is in accordance with Subsections IWB, IWC, and IWD, respectively.

The control rod drive mechanism (CRDM) thermal sleeves are examined under an augmented ISI inspection program. The scope of examination is to ultrasonically test (UT) the five (5) thermal sleeves with the worst wear on each unit. The plan for managing thermal sleeve wear is to obtain measured (UT) wear data points on each unit at the five (5) designated thermal sleeve reactor core locations during three (3) different outages. The frequency for inspection of the reactor vessel head thermal sleeve for loss of material due to wear will be re-evaluated after the accumulation of the three (3) data points on each of the five (5) designated thermal sleeves. The three (3) series of examinations will be performed prior to the period of extended operation. Subsequently, the required frequency for further inspections, if required, will be determined using the guidance provided in WCAP-16911-P, "Reactor Vessel Head Thermal Sleeve Wear Evaluation for Westinghouse Domestic Plants."

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program implements the required component examination schedule in accordance with ASME Section XI, Subsection IWB-2400, IWC-2400 or IWD-2400 and examination categories, applicable components, examination methods, acceptance standards, and frequency of examination as specified in Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1. The examination methods specified in Tables IWB-2500-1, IWC-2500-1 and IWD-2500-1 are based on approved industry standards for detecting age-related degradation of components. The program requires that indications and relevant conditions detected during examinations be evaluated in accordance with ASME Section XI, Articles IWB-3000 for Class 1, IWC-3000 for Class 2, and IWD-3000 for Class 3. The program directs that repair and replacement activities be performed in conformance with IWA-4000. This condition-monitoring program provides adequate monitoring methods that are effective in detecting the relevant aging effects and the frequency of monitoring is adequate to prevent significant age-related degradation.

In accordance with 10 CFR 50.55a(g)(4)(ii), the ISI program is updated each successive 120-month inspection interval to comply with the requirements of the latest edition of the ASME Code specified twelve months before the start of the inspection interval.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program includes all component inspection activities required by ASME Code, Section XI, Subsections IWB, IWC, and IWD in conjunction with component types that are covered by the following license renewal aging management programs as described within the referenced aging management program bases documents listed below:

- Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components, B.2.1.5
- Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS), B.2.1.6
- PWR Vessel Internals, B.2.1.7
- Steam Generators, B.2.1.10
- One-Time Inspection of ASME Code Class 1 Small-Bore Piping, B.2.1.22