NRR-DMPSPEm Resource

From:	Haskell, Russell
Sent:	Friday, December 22, 2017 3:09 PM
To:	'Mitchel.Mathews@exeloncorp.com'
Subject:	Dresden Nuclear Power Station, Units 2 and 3 - RAIs to Support Request to Revise TS 5.5.12 "Primary Containment Leakage Rate Testing Program" (EPID: L-2017-LLR-0228)
Attachments:	FINAL - DNPS, Unit 2 and 3 - RAIs for ILRT Extension Request.pdf; Revised PRA RAI 9.pdf

Dear Mr. Mathews,

By letter dated May 3, 2017, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17123A104), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request for Dresden Nuclear Power Station, Units 2 and 3 (DNPS). The amendment proposes to revise existing DNPS Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

On December 8, 2017, the NRC provided EGC with a "DRAFT" set of RAIs regarding the ILRT extension request. On December 20, 2017, EGC and support personnel held a teleconference with NRC staff to clarify several of these RAIs.

Based on the teleconference:

- PRA-RAI-7.a - Class II Decay Heat Removal - **DELETED**
- PRA-RAI-7.b - Class II Decay Heat Removal - **DELETED**
- PRA-RAI-8 - Frequency of EPRI Category 3b - **DELETED**
- PRA-RAI-9.a - Overall Baseline LERF Risk Assessment – Uncertainty - **REVISED** (see attached)*

The attached RAIs are determined to be final. As such, and as discussed during the call with you, the NRC staff is requesting EGC's response to these RAIs by January 19, 2018.

If you have any questions, please contact me at (301) 415-1129 or by email at Russell.Haskell@nrc.gov.

Sincerely,

Russell S. Haskell II
United States Nuclear Regulatory Commission (NRC)
Licensing Project Manager - NRR/DORL/LPL 3
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Hearing Identifier: NRR_DMPS
Email Number: 70

Mail Envelope Properties (DM5PR09MB1194537AE1F444C1AE4658F4ED020)

Subject: Dresden Nuclear Power Station, Units 2 and 3 - RAIs to Support Request to Revise TS 5.5.12 "Primary Containment Leakage Rate Testing Program" (EPID: L-2017-LLR-0228)
Sent Date: 12/22/2017 3:08:53 PM
Received Date: 12/22/2017 3:08:54 PM
From: Haskell, Russell

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Tracking Status: None

Post Office: DM5PR09MB1194.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	1698	12/22/2017 3:08:54 PM
FINAL - DNPS, Unit 2 and 3 - RAIs for ILRT Extension Request.pdf	155752	
Revised PRA RAI 9.pdf	88530	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST – REGARDING

EXTENSION OF THE 10 CFR 50 APPENDIX J

CONTAINMENT TYPE A & TYPE C TEST INTERVALS

DRESDEN NUCLEAR POWER STATION UNITS 2 & 3

DOCKET NOS. 50-237 & 50-249

By letter dated May 3, 2017, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17123A104), Exelon Generation Company, LLC, (the licensee) submitted a license amendment request to revise Dresden Nuclear Power Station, Unit 2 and 3 (DNPS) Technical Specification 5.5.12. The proposed amendment would extend the existing Type A integrated leakage rate test program test interval from 10 years to 15 years, and to adopt an extension of the containment isolation valve leakage rate testing (i.e., Type C test) frequency from 60 months to 75 months. The additional information below is needed to support the NRC staff's continued technical review of the license amendment request (LAR).

PRA-RAI-1 Clarification of Probabilistic Risk Assessment (PRA) Model Version

In Table 5.1-2, "Accident Class 7 Failure Frequencies and Population Doses (DNPS Base Case Level 2 Model)," of Attachment 3 of the LAR (page 5-8), there is a reference to a 2014 PRA model in the column titled "2014 PRA Release Frequency / Yr." However, the 2013A PRA model appears to be used for the PRA evaluation supporting the LAR.

The NRC staff requests the licensee either correct the model referenced or justify the applicability of the model, as referenced in the LAR.

PRA-RAI-2 Technical Adequacy of the Internal Flooding PRA Model

In Section 3.4.2.5, "Consistency with Applicable PRA Standards," of Attachment 1 of the LAR (page 21 of 78), includes two focused scope peer reviews of the Internal Flooding PRA (IFPRA) model, one dated March 2009 and the other dated May 2010. In Section A.2.4, of Attachment 3 (page A-5) only discussed the March 2009 peer review, while in Table A-1, "2010 Focused Area IF Peer Review Findings/Status and Impact to Application," of Attachment 3 (beginning on page A-9) appears to provide the Findings and & Observations (F&Os) and associated resolutions from the May 2010 peer review. Furthermore, Section 3.4.2.5 identifies both of these peer reviews as having been done using NRC Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 (ADAMS Accession No. ML070240001). Section 3.4.2.5 states that an independent PRA peer review of the internal events PRA (IEPRA) was conducted in 2016 using RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), and that all supporting requirements (SRs) were included except those related to internal flooding (which were previously peer reviewed in 2009).

The NRC staff uses Revision 2 of RG 1.200, to assess the technical adequacy of the PRA used to support risk-informed LARs. Therefore, the NRC staff requests that the licensee:

a. Provide a gap assessment of the IFPRA model against RG 1.200, Revision 2 in regards to this LAR.

b. Provide justification why the IFPRA F&Os from the 2009 peer review were not provided in the LAR, or provide these F&Os to the NRC staff for review.

<u>PRA-RAI-3</u> Resolution of PRA Peer Review F&Os

In Appendix A, "PRA Technical Adequacy," Table A-2, "2016 FPIE Peer Review Findings and Impact to Application," of Attachment 3 (beginning on page A-17), the licensee presents the F&Os from the 2016 peer review of the IEPRA along with their resolutions for this LAR.

a. F&O 2-12 (pg. A-22) pertaining to large early release frequency (LERF), states that credit for mitigating actions by operators, fission product scrubbing, and expected beneficial failures in significant accident progression sequences are included in the LERF analysis, but the licensee provided no justification for equipment survivability and successful human actions under adverse environments or after containment failure. In addition, the resolution states that Capability Category I (CC-I) is met.

   The NRC staff requests that the licensee explain how CC-I is satisfied.

b. F&O 5-1 (pg. A-25) pertains to the modeling of human reliability analysis (HRA) dependencies. The peer review team found that the "Dresden PRA model does not account for the influence of success or failure in preceding human actions and system performance on the human event under consideration," including timing, factors that could lead to dependence, and availability of resources and therefore SRs HR-G7 and QU-C2 were found Not Met. In the "Impact to Application" column of the table, the licensee however states that the HRA dependency methodology is considered adequate for this application.

   The NRC staff requests that the licensee either provide the results of a focused scope peer review on the HRA dependency analysis to demonstrate that SRs HR-G7 and QU-C2 are met, or, alternatively provide the results of a sensitivity study, which applies the minimum joint human error probability (HEP) floor value of 1E-6 for internal event PRAs.

In Section 3.4.2.5, Table A-2, of Attachment 3, the licensee stated that five SRs were assessed as not applicable to the DNPS PRA. The results of that assessment are used as the basis for the capability assessment provided in the LAR.

c. The NRC staff requests the licensee to provide greater specificity regarding the five SRs mentioned in the above reference and their relevance to this LAR.

<u>PRA-RAI-4</u> PRA Differences

In Section 4.2, "Plant Specific Inputs," of Attachment 3 (page 4-8), the licensee states that no substantive differences exist between DNPS, Unit 2 and Unit 3 that are judged to affect the conclusions of the PRA, as such, no separate PRA quantification is conducted for DNPS Unit 3.

The NRC staff requests that the licensee:

a. Provide clarification as to whether this conclusion also applies to the IFPRA model, and, if not, discuss the impact of any risk-significant differences.

b. Provide an explanation of how the results from the DNPS, Unit 2 PRA model are sufficient for assessing dual unit risk for this LAR.

PRA-RAI-5 Accident Sequence Category Assumptions

In Table 4.2-5, “Accident Sequence Category Descriptions from the License Renewal Severe Accident Management Alternatives Evaluation,” of Attachment 3 (page 4-20), the licensee indicated that the “time of end of release” for consequence category L2-10 (intact containment) is 36 hours. This assumption is critical to the determination of the estimated change in population dose for the extended ILRT interval application.

The NRC staff requests the licensee to explain the likelihood of further releases after 36 hours, and if likely, provide an updated assessment of the releases against the population dose criteria for internal as well as external events.

PRA-RAI-6 Containment Accident Pressure (CAP) Credit

In Section 3.3.3, “DNPS 10 CFR 50, Appendix J, Option B Licensing History,” Table 3.4.1-1, of Attachment 1 (page 13 of 78), the licensee states that it relies upon containment over-pressure for emergency core cooling system (ECCS) performance and for pressure effects on net positive suction head (NPSH) for the low pressure core injection (LPCI) and condensate spray (CS) pumps, and that details are provided in Section 3.2, of Attachment 1 of the LAR. In Section 5.8, “Containment Overpressure Impacts on CDF,” of Attachment 3 (pages 5-36 and 5-37), provides an estimate of the increase in internal events core damage frequency (CDF) by applying the Event Class 3b probability of large preexisting containment leakage at 15 years to the containment isolation failure probability in the PRA model.

It is not clear to the NRC staff how increasing the containment isolation failure probability accounts for the increase in CDF for accident scenarios that credit NPSH for success, including the following example accident scenarios identified in Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals”:

- Loss of coolant accident (LOCA) scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection, and

- Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside containment.

The NRC staff requests that the licensee:

a. Identify all accident sequences, and provide a brief description of each, that rely on NPSH to prevent core damage. Describe how loss of containment accident pressure (CAP)-related NPSH is included in the PRA model and evaluated for CDF and LERF for these sequences. If only increasing containment isolation failure probability is assessed, provide justification.

b. Confirm that the PRA model used for the CAP risk assessment included all initiating events or conditions which correspond to the two EPRI guidance general event categories.

c. Describe thermal hydraulic analyses performed in support of loss of CAP evaluation, and how they are credited in the PRA if applicable.

d. If containment cooling is credited for NPSH concerns, explain how this credit is taken in the PRA model given the assumption of a pre-existing leak.

e. It appears that CAP-related risk does not include external events contribution. Include CAP-related risk in the evaluation of external events risk, and describe the approach.

f. Evaluate the CAP-related LERF contribution and describe the method used. If a PRA model is used, provide justification for credits which result in a reduction of LERF risk. If a PRA model was not used, confirm that the EPRI Report No. 1009325 allowed first order approximation method of assuming that the change in CDF equals the change in LERF. If a different method had been taken, describe and provide justification.

PRA-RAI-7 (Formerly PRA-RAI-9) Overall LERF Risk Assessment

In Table 5.7-5, "DNPS External Events Contributor Summary," of Attachment 3 (page 5-53), the following data is depicted in the last two rows:

	CDF(1/yr)	**LERF(1/yr)**
Total for External Events	4.98E-5	8.58E-6
Internal Events	3.22E-6	7.31E-7

Summing external and internal frequencies results in a total CDF of 5.30E-5/yr and a total LERF of 9.21E-6/yr. The total LERF is close to the RG 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, (ADAMS Accession No. ML100910006), guideline of 1E-5/yr.

The NRC staff requests that the licensee provide additional information regarding the assumptions and uncertainties in its estimate:

a. Assess quantitatively or qualitatively the key assumptions or sources of uncertainty on the total LERF to include the following sources and provide a discussion of the assessment results:

1. Fire contribution,
2. CAP-related contribution,
3. Seismic contribution,
4. Steel liner corrosion contribution, and
5. Class II decay heat removal contribution.

b. With regard to fire-related risk, in Section 5.7.1, of Attachment 3, explains that the DNPS Individual Plant Examination of External Events (IPEEE) fire analysis was used in the risk assessment of the extended ILRT interval because a fire PRA model was not available. Given that the IPEEE analysis was completed in March 2000, provide a discussion of the impact on the risk assessment results of the current plant configuration and operating experience including a discussion on plant changes that have been made since the IPEEE that would significantly increase/decrease fire risk, including for screened fire areas.

c. With regard to CAP-related risk, identify any key assumptions or sources of uncertainty for the loss of CAP evaluation, given the assumed pre-existing leak, and discuss the assessment of them.

d. Identify and discuss any other sources of uncertainty which could be important for assessing the total LERF, and include them in the evaluation.

e. Assess the cumulative effect of these considerations and provide justification for why the total LERF is consistent with the RG 1.174 guidance when LERF-related assumptions or uncertainties are considered.

PRA-RAI-8 (Formerly PRA-RAI-10) Total Increase in CDF and LERF

The NRC staff requests the licensee to provide the total increase in CDF and LERF for internal and external events, including the CAP related risk increase.

SBPB-RAI-1

The NRC staff notes that the licensee adopted Option B of 10 CFR 50, Appendix J following the January 11, 1996, issuance of Amendment No. 148 to Facility Operating License No. DPR-19 and Amendment No. 142 to Facility Operating License DPR-25 for DNPS, Units 2 and 3, respectively. Per the guidance of NEI 94-01 Revision 0, Section 10.2.3.2 and subject to the four provisions identified in Regulatory Position “C” of Regulatory Guide 1.163, both DNPS Unit 2 and Unit 3 are currently allowed to extend the test intervals for Type C CIVs up to 60 months.

In Section 10.2.3.2 of NEI 94-01 of both Revision 0 and Revision 3-A reads in part:

> Test intervals for Type C valves may be increased based upon completion of two consecutive periodic As-found Type C tests where the result of each test is within a licensee's allowable administrative limits.

In Section 3.5.5, “Type B and Type C Local Leak Rate Testing Program Implementation Review,” of Attachment 1 (page 55 of 78), the NRC staff notes that a significant minority (i.e., approximately 45%) of the population of both DNPS, Unit 2 and Unit 3 Type C tested components (CIVs) are not on an extended frequency of 60 months. The ability to extend the Type C test intervals from 60 months to 75 months would be supported by having a large percentage of CIVs already qualifying for 60 months with some margin.

The NRC staff requests that the licensee provide additional information about regarding Sections 3.5.4 and 3.5.5 of the LAR. In particular, do the Type C test results indicate that a large percentage of valves have not successfully made it to a 60 month test interval frequency?

SBPB RAI-2

DNPS TS 5.5.12a indicates that the leakage rate testing program will be in accordance with Regulatory Guide 1.163 which endorsed NEI 94-01. Determining the As-found minimum pathway combined Type B and C test totals verifies that the requirement had been met at all times when containment integrity was required. In approving the industry proposal to allow ILRT intervals to routinely extend to a maximum of 15 years, the NRC staff required a change to the NEI 94-01 guidance such that NEI 94-01 Revision 2-A and Revision 3-A state that acceptance criteria for the combined As-found leakage rate for all penetrations subject to Type B or Type C testing “be less than $0.6L_a$” rather than “recommended” to be less than 0.6 L.

In Table 3.5.4-1, “DNPS, Unit 2 Types B and C LLRT Combined As-Found/As-Left Trend Summary,” of Attachment 1 (page 54 of 78), the aggregate LLRT values in row: “AF Min Path (scfh),” displays outage totals as “Fraction of L_a.” The NRC staff notes that for refueling outage D2R22 in 2011 and that for refueling outage D2R24 in 2015, the as found minimum pathway that “Fraction of L_a “ values were recorded at 0.928 and 0.701, respectively. In both instances, there is neither an explicit indication in the LAR that this represented a failure to meet the surveillance performance criterion nor that past operability of the primary containment had been evaluated given the As-found Type B and C total exceeding 0.6 L_a. Both occurrences represent a prima facie entrance into the margin reserved for ensuring the overall primary containment performance criterion of L_a had not been challenged given that the combined Type B and C leakage did not necessarily account for all containment leakage potential.

The NRC staff requests the licensee provide additional information:

a. For example, corrective actions

b. For improvements to the leak testing program, etc., that justify the requested Unit 2 Type A and Type C test extensions during each of the last three refueling outages, D2R22, D2R23 (in 2013), and D2R24

c. For the combined <u>As-found</u> Type B and C minimum pathway test results that exceeded $0.6L_a$ by significant margins

PRA-RAI-9 (revised)

PRA-RAI-9 Overall ~~Baseline~~ LERF Risk Assessment

Table 5.7-5 "DNPS External Events Contributor Summary," in Attachment 3 (page 5-53), the following data is depicted in the last two rows:

	CDF(1/yr)	**LERF(1/yr)**
Total for External Events	4.98E-5	8.58E-6
Internal Events	3.22E-6	7.31E-7

Summing external and internal frequencies results in a total CDF of 5.30E-5/yr and a total LERF of 9.21E-6/yr. The total LERF is close to the RG 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, (ADAMS Accession No. ML100910006), guideline of 1E-5/yr.

The NRC staff requests that the licensee provide additional information regarding the assumptions and uncertainties in its estimate:

a. Assess quantitatively or qualitatively the key assumptions or sources of uncertainty on the total ~~baseline~~ LERF to include the following sources and provide a discussion of the assessment results:

1. Fire contribution,
2. CAP-related contribution,
3. Seismic contribution,
4. Steel liner corrosion contribution, and
5. Class II decay heat removal contribution.

b. With regard to fire-related risk, LAR Attachment 3, Section 5.7.1 explains that the DNPS Individual Plant Examination of External Events (IPEEE) fire analysis was used in the risk assessment of the extended ILRT interval because a fire PRA model was not available. Given that the IPEEE analysis was completed in March 2000, provide a discussion of the impact on the risk assessment results of the current plant configuration and operating experience including a discussion on plant changes that have been made since the IPEEE that would significantly increase/decrease fire risk, including for screened fire areas.

c. With regard to CAP-related risk, identify any key assumptions or sources of uncertainty for the loss of CAP evaluation, given the assumed pre-existing leak, and discuss the assessment of them.

d. Identify and discuss any other sources of uncertainty which could be important for assessing the total ~~baseline~~ LERF, and include them in the evaluation.

e. Assess the cumulative effect of these considerations and provide justification for why the total ~~baseline~~ LERF is consistent with the RG 1.174 guidance when LERF-related assumptions or uncertainties are considered.

I am teleworking today, so I will not be at my desk.