From:	Marshall, Michael
To:	[Licensee] Ron Reynolds (Exelon)
Cc:	Wolniak, Denise J:(Exelon Nuclear); James Danna (James.Danna@nrc.gov)
Subject:	NINE MILE POINT, UNIT 2 – DRAFT REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO INCREASE ALLOWABLE MSIV LEAKAGE RATES (L-2019-LLA-0115)
Date:	Friday, February 14, 2020 10:19:00 AM

Hello Ron,

By letter dated May 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19151A537), as supplemented by letter dated November 21, 2019 (ADAMS Accession No. ML19325D201), Exelon Generation Company, LLC (Exelon) requested that the U.S. Nuclear Regulatory Commission (NRC) amend the Technical Specifications, Appendix A of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2).

Exelon's proposed license amendment request (LAR) would change Technical Specification Surveillance Requirement 3.6.1.3.12 for the main steam isolation valve (MSIV) leakage rate. The current leakage rate limit of less than or equal to 24 standard cubic feet per hour (scfh) when tested at greater or equal to 40 pounds per square inch gauge (psig) for each MSIV would be revised to allow a leakage rate of less than or equal to 50 scfh when tested at greater or equal to 40 psig for each MSIV. Exelon stated that the changes to the leakage rate limits are based on a revised radiological analysis of the design basis loss of coolant accident in accordance with the alternative source term (AST) methodology. Additionally, the LAR describes a revised radiological analysis for environmental qualification and vital area access, which are based on the Technical Information Document -14844 source term methodology.

The NRC staff has determined that additional information is needed to complete its review of the request. The additional information is needed to complete the review and to determine whether the NRC regulatory requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident source term" and 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19, "Control Room," are met.

The NRC staff has reviewed the information provided in the LAR and has determined that additional information is needed to complete its review. Below is the NRC staff's request for additional information. The request for additional information was discussed with you on February 14, 2020, and it was agreed that Exelon's response would be provided within 90 days from the date of this email.

The RAIs listed below are not a complete listing of the additional information needed to complete the NRC staff's review. RAIs 1 to 3 were provided in a separate email dated October 23, 2019 (ADAMS Accession No. ML19296A186).

<u>Background</u>

Exelon's revised loss of coolant accident (LOCA) radiological analysis (Calculation No. H21C-106, Revision 3), contained in Enclosure A of the LAR, proposes to modify several assumptions and inputs previously used to model the main steam isolation valve (MSIV) leakage pathway after a postulated design basis LOCA.

As stated in NRC Regulatory Issue Summary 2006-04, “Experience with Implementation of Alternative Source Terms,” dated March 7, 2006 (ADAMS Accession No. ML053460347), any licensee who chooses to reference the assumptions in AEB 98-03 should provide an appropriate justification that the assumptions are applicable to their particular design.

Section 50.67 of 10 CFR requires, in part, that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE), (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE, and (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

Appendix A to 10 CFR Part 50, GDC 19, requires, in part, that the control room be maintained in a safe, habitable condition under accident conditions by providing adequate protection from a dose that would not exceed 5 rem TEDE for the duration of the accident.

SRP 15.0.1, states, in part that:

> The models, assumptions, and parameter inputs used by the licensee should be reviewed to ensure that the conservative design basis assumptions outlined in RG [regulatory guide] -1.183 have been incorporated. These assumptions provide an integrated approach to performing the individual analyses and licensees are generally expected to use these assumptions or to propose acceptable alternatives. Licensee-proposed alternatives to this guidance may be accepted if technically appropriate and of an appropriate level of conservatism. Significant departures from this guidance will warrant additional review.

<u>Draft Requests for Additional Information</u>

4. In Attachment 1, page 5 of the LAR the licensee states:

> The 20-group probabilistic distribution methodology has been previously approved at Clinton (Reference 10), Limerick (Reference 11), and LaSalle (Reference 12) [Adams Accession Nos. ML052570461, ML062210214, and ML101750625, respectively].

The NRC staff notes that the cited precedents included a ruptured main steam line (MSL) to maximize the dose consequences from MSIV leakage. Appendix A of AEB-98-03 included this assumption as shown below:

> The staff's well-mixed deposition model assumes that each segment of piping in the RADTRAD nodalization is well-mixed. The unbroken

> main steam lines in the RADTRAD nodalization are modeled as two segments. The first segment is the length of piping between the reactor vessel and the first MSIV. The second segment is the length of piping between the first MSIV and the second MSIV. The broken main steam line is modeled as one segment of piping. This segment is the length of piping between the first MSIV and the second MSIV.

The licensee addressed this issue in Attachment 1, page 8 of the LAR which states:

> All MSLs in the MSIV leakage release pathways are seismically designed and supported to withstand the Safe Shutdown Earthquake (SSE) and thereby comply with RG 1.183, Appendix A, Section 6.5 requirement. The recirculation line break is the limiting event for fuel failure. It is not credible to assume two initiating limiting events, a recirculation line break and a break on the main steam line in a single design basis event.
>
> All four MSL headers are Seismic I and QA Cat 1 from the RPV nozzle to seismic boundary break at the TSV [turbine stop valve]; therefore, they are qualified to withstand the SSE, and they comply with the RG 1.183, Appendix A, Section 6.5 requirement to be credited for aerosol deposition. Therefore, the MSIV leakage pathway boundary is extended up to the TSV.

The NRC notes that while it is true that mechanistically a recirculation line break would be expected to present a more significant challenge to the reactor core than a ruptured MSL, the source term used to satisfy 10 CFR 50.67 is a deterministic source term imposed on the facility to test the ability of systems to mitigate the releases sufficiently to meet predetermined acceptance criteria. Assuming a ruptured MSL in the evaluation of the acceptability of MSIV leakage criteria fulfills the underlying guidance from RG 1.183 that assumptions should be selected with the objective of maximizing the postulated radiological consequences.

The NRC staff notes that the integrity of the entire reactor coolant pressure boundary must comply with SSE requirements to satisfy Appendix A to Part 100. The assumption of a ruptured MSL for evaluating MSIV leakage in conjunction with a deterministic source does not imply a ruptured MSL in addition to a recirculation line rupture. Rather the evaluation assumes a ruptured MSL (with a deterministic source term) to maximize the dose contribution from MSIV leakage.

Please provide additional information to justify that assuming a recirculation line rupture instead of a main steam line rupture is consistent with the guidance from RG 1.183 that assumptions should be selected with the objective of maximizing the postulated radiological consequences.

5. The LAR discusses the licensee's review of various NRC AST safety evaluation (SE) and how these SEs identified the staff's concern with how much deposition is assumed in the LOCA MSIV leakage pathways when using the AEB-98-03 model, "Assessment of the Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998 (ADAMS Accession No.

ML011230531).

In the NRC staff's SE dated May 29, 2008 (ADAMS Accession No. ML081230439), to approve Exelon's full implementation of the AST methodology for Nine Mile Point 2, the NRC staff indicated that it had concerns regarding the use of AEB 98-03. At that time, the NRC staff based its approval of the LAR, in part, upon additional conservatism in the deposition model used. Specifically, the SE, in part, states:

However, for additional conservatism, and to address [NRC] concerns historically documented by the NRC staff, the licensee used [1/2 of] the 3rd percentile settling velocity of 0.000066 m/sec. The NRC staff agrees that the 3rd percentile conservatively reflects the effectiveness of drywell spray activity removal in containment upstream of this pathway.

The NRC staff notes that the current licensing basis for Nine Mile Point 2 provided in the supporting calculation previously transmitted to the NRC (see Calculation No. H21C-106, Revision 0 (ADAMS Accession No. ML071580354)), page C2 indicates that ½ of the 3rd percentile is equivalent to the settling velocity of 0.000066 m/sec.

In Attachment 1, page 5 of the LAR the licensee states:

> The revised LOCA dose analysis implements a 20-group probabilistic settling velocity distribution for MSIV leakage rather than using the AEB-98-03 single, median value, model. The 20-group probabilistic distribution methodology has been previously approved at Clinton (Reference 10), Limerick (Reference 11), and LaSalle (Reference 12). The same settling velocity probability distribution function shown in Equation 5 of AEB-98-03 is used to conservatively calculate aerosol settling velocity as follows […]

The NRC staff notes that the analyses cited as precedents did not credit drywell sprays. Page 96 of NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," (ADAMS Accession No. ML063480542), provides details on how sprays impact aerosols. NUREG/CR-5966 indicates that the sprays shift the sizes of aerosols in the containment towards those that are removed most slowly (the mean aerosol size decreases as the sprays operate). The licensee's estimates of aerosol deposition in the steam lines is determined using, in part Equation 5 of AEB 98-03. Equation 5 of AEB 98-03 provides the aerosol settling (and thus the aerosol deposition) in the steam line and indicates that the aerosol settling is proportional to the square of the diameter of the aerosols. Because the sprays shift the size of the aerosols to smaller sizes, the aerosols settling in the steam lines would decrease due to these smaller diameter aerosols.

In the 2007 Nine Mile Point 2 LAR to incorporate 10 CFR 50.67 into the Nine Mile Point 2 licensing basis, Calculation H21C-106, Revision 0, page C1 discusses a "penalty" on the sedimentation velocity (or aerosol settling velocity) used for bypass pathways to account for the recognition that the sprays preferentially remove large particles in primary containment.

As discussed in Nine Mile Point 2's safety evaluation dated May 29, 2008, the NRC staff

stated that they had issues with the use of AEB 98-03 for modeling aerosol deposition for Nine Mile Point 2. In this safety evaluation the staff stated that the licensee used a settling velocity of 0.000066 m/sec to address the staff's issues regarding the use of AEB 98-03 and that this value was sufficiently conservative (along with other conservatisms) to reflect the effectiveness of the sprays.

From an examination of the submitted information it appears that the licensee considers the aerosol removal by sprays and aerosol removal in the main steam lines as independent removal mechanisms. The NRC staff notes that regardless of the specific removal mechanisms involved, larger aerosol particles in the containment atmosphere will be the preferentially removed therefore making subsequent removal by deposition in downstream piping more challenging.

Based upon the above observations, it is unclear: 1) why assuming that the aerosol deposition in the steam line is independent of the RHR drywell spray credit, and 2) how input parameters to the 20-group method reflect changing aerosol characteristic due to the drywell sprays.

Please provide technical information to:

a) Describe how the gravitational settling credited in the main steam lines, using the 20-group method, considers the changing aerosol characteristics (i.e., aerosol size and density distributions) due to the sprays and as these aerosols move through the main steam lines.

b) Explain why the results of the 20-group method when crediting sprays are valid for Nine Mile Point 2.

6. Assumption 6.5 in Appendix A of RG 1.183 provides guidance on an acceptable model for crediting the deposition of elemental iodine in the main steam piping downstream of the MSIVs, and states that the amount of reduction allowed will be evaluated on an individual case basis. Assumption 6.5 refences the J.E. Cline model (ADAMS Accession No. ML003683718). On page 10 of Attachment 1 of the LAR it states that the J.E. Cline methodology is used to calculate the time-dependent deposition and resuspension rates of elemental iodine for the MSIV release pathways in the revised LOCA analysis. The LAR proposes to increase the elemental iodine deposition in the steam line which would be expected to decrease the estimated calculated doses from the postulated LOCA.

The J.E. Cline model describes that the steam line temperatures assumed after the postulated LOCA directly impact the amount of elemental deposition credited in the steam line, which in turn, directly impact the estimated calculated doses from the postulated LOCA. As described in Calculation H21C-106, Revision 3, page 55, steam line temperatures are assumed from the J.E. Cline model. No justification for why these steam line temperatures are applicable for Nine Mile Point 2 is provided.

In addition, Attachment 1 of the LAR, page 8, states the steam line piping steel heat-up due to fission product deposition in the steam line is conservatively estimated to be 0.5 degrees Fahrenheit per hour (°F/hr) based upon Reference 14 ("BWR Steam Line Radionuclide Concentration Distribution following a DBA LOCA" (ADAMS Accession No. ML102380174) of the LAR. The NRC staff notes that Reference 14 is not part of the

J.E. Cline model or RG 1.183, Revision 0. Therefore, the NRC staff reviewed the assumptions used to derive the estimated heat up in the steam line and the information regarding the steam line temperatures provided in the LAR.

a) The heat-up values calculated in Reference 14 appear to range between 0.5 °F/hr and 2.5 °F/hr (see Section 3.2.2, page 32 of the LARs Reference 14) depending upon the assumptions made. No justification is provided on why a heat up rate of 0.5 °F /hr is applicable to Nine Mile Point 2. Please provide additional information to justify the use of the heat up rate of 0.5 °F /hr.

b) Per Reference 14 of the LAR, the piping heat up value of 0.5 °F/hr is based, in part, upon only the "Group 2 "(Cesium Iodide and Rubidium Iodide) radionuclides, however, the LAR credits deposition of a significant amount of other radionuclides (that include almost all the aerosols and a significant amount of the elemental iodine that leaks into the steam lines). It is not clear why only Group 2 radionuclides are considered and not considering the other radionuclides could underestimate the decay heat, steam line temperatures, and estimated doses. Please provide additional information to explain why use of only Group 2 radionuclides does not result in an underestimation of the decay heat, steam line temperatures, and estimated doses.

   The value of 0.5 °F/hr is based, in part, upon the assumption that a quarter of the deposited power would escape based upon unattenuated gamma radiation. However, it appears that the decay power is based upon the thermal power. No information is provided as to why assuming a quarter of the deposited power is lost, due to gamma radiation, is an appropriate assumption that would result in an accurate heat up rate or estimated doses. Also, the value of 0.5 °F/hr is based, in part, upon on the assumption that the amount of Group 2 (i.e., CsI) mass leaked to the environment (via the steam line pathway) by the end of "the time frame of interest" is about 2.3E-5 (as a fraction of core inventory). No justification is provided on the basis of this value and how the "time frame of interest" is defined.

   Please provide additional information to:

   i. Explain why assuming a quarter of the deposited power is lost, due to gamma radiation, results in an accurate heat up rate and estimated doses.

   ii. Clarify how the value of 2.3E-5 relates to the deposition in the steam line and state how the "time frame of interest" is defined.

c) The value of 0.5 °F/hr heat up rate assumes that the deposited power would uniformly heat up steam piping that is 5 centimeter (cm) thick. However, the heat up rate would not be uniform because the radionuclides would be deposited on the inner surfaces of the pipe (where the temperatures would be higher than other portions of the pipe). Also, the assumed 5 cm pipe thickness in Reference 14 conflicts with the pipe thickness of 2.5 cm (as shown in Table 3 of the J.E. Cline model) and used to create the steam line temperature profile used by Nine Mile Point 2 in the revised LOCA analysis. No information is provided as to why the

Reference 14 assumptions of a uniform heat-up rate and 5 cm pipe thickness appropriately reflect the expected temperatures in the steam line piping when the temperatures in the piping would not be uniform and the temperatures assumed by Nine Mile Point 2 are derived upon different main steam pipe thicknesses. Please provide additional information to justify the assumption of a uniform heat up rate and a 5 cm pipe thickness.

d) In Attachment 1, page 8, the rise in steam line temperature (due only to fission product deposition) during the accident period of 720 hours is 360 °F (assuming no radioactive decay and heat loss). However, it appears that in Table 5 of Calculation H21C-106, Revision 3, the steam line temperature used to determine the revised elemental deposition is decreasing. Using the equation on page 56 of Calculation H21C-106, Revision 3, the steam line temperature would continue to be about 80 °F at 720 hours which is below the temperature rise value of 360 °F due to only deposition.

Calculation H21C-106, Revision 3, page 21, states that:

> The inboard piping is connected to the RPV [reactor pressure vessel] and subjected to achieve the temperature the same as the RPV dome prior to water being restored around 1 hr. Preliminary results using MAAP for Quad Cities indicates that the temperatures in the RPV head may briefly spike over 700°F but then fall below 600 °F. The temperature in the first MSL [main steam line] node also exceeds 600 °F in a few cases for short duration, but generally stays below 600 °F. In the worst case the temperature transient in the inboard piping may last less than an hour that may potentially impact the aerosol physics and plateout mechanism, which may affect the aerosol removal credited in the analysis.

As discussed above, the steam line temperatures impact the amount of elemental deposition, thereby, impacting the doses. The temperature profile used for calculating the elemental deposition in the revised LOCA analysis does reflect any increasing steam line temperatures as indicated in Reference 14. Not considering these increases in steam line temperature would underestimate the dose results.

Please provide additional information to justify use of:

i. the steam line temperature profile proposed in the LAR is applicable to Nine Mile Point 2,

ii. the Reference 14 analysis are correct and applicable to Nine Mile Point 2,

iii. the preliminary MAPP results for Quad Cities are appropriate for design basis calculations for Nine Mile Point 2.

Sincerely,
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Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410