From:	Lee, Samson
To:	Loomis, Thomas R:(Exelon Nuclear)
Subject:	FitzPatrick request for additional information: License Amendment Request for Application of the Alternative Source Term for Calculating Loss-of-Coolant Accident Dose Consequences (EPID: L-2019-LLA-0171 and L-2019-LLE-0020)
Date:	Monday, February 03, 2020 2:57:00 PM
Attachments:	ARCB RAI for JAFNPP AST_Feb 3 2020.docx

By letter dated August 8, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19220A043), Exelon Generation Company, LLC (Exelon, the licensee), submitted a license amendment request (LAR) for adopting the Alternative Source Term (AST), in accordance with 10 CFR 50.67, for use in calculating the Loss-of-Coolant Accident (LOCA) dose consequences at the James A. FitzPatrick Nuclear Power Plant. The NRC staff has reviewed the LAR and determined that additional information is required to complete the review. The NRC staff's requests for additional information (RAIs) are attached. These RAIs are in the accident dose areas. The Exelon staff indicated that a clarification call was not necessary and there was no proprietary or sensitive information. The Exelon staff requested, and NRC agreed, to a RAI response by April 3, 2020.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. Please note that if you do not respond to this request by the agreed-upon date or provide an acceptable alternate date, we may deny your application for amendment under the provisions of Title 10 of the Code of Federal Regulations, Section 2.108. If circumstances result in the need to revise the agreed upon response date, please contact me at (301) 415-3168 or via e-mail Samson.Lee@nrc.gov.

OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION
RELATED TO THE LICENSE AMENDMENT REQUEST REGARDING
IMPLEMENTATION OF THE ALTERNATIVE SOURCE TERM FOR THE
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333

BACKGROUND:

By letter dated August 8, 2019 Agencywide Documents Access and Management System (ADAMS) Accession No. ML19220A043, Exelon Generation Company, LLC (EGC) (the licensee), requested a license amendment to fully implement an alternative source term (AST) methodology at James A. FitzPatrick Nuclear Power Plant (JAFNPP). The proposed license amendment request (LAR) would make the following changes to the current licensing and design basis for JAFNPP: revisions to several Technical Specifications (TS) and associated Bases to reflect implementation of AST methodology in the design basis dose consequence loss of coolant accident (LOCA), deletion of the Main Steam Leakage Collection (MSLC) system TS and associated Bases, revision to increase the allowable TS leakage for the Main Steam Isolation Valves (MSIVs), revision of the Standby Liquid Control (SLC) system TS and associated Bases, and revision of the Ventilation Filter Testing Program (VFTP) TS.

The licensee stated in the LAR that the revised LOCA radiological analysis is performed using the AST methodology, established as the licensing basis for this accident, and that 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), Criterion 19—Control room, guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (ADAMS Accession No. ML003716792), guidance in Standard Review Plan (SRP) 6.5.2, "Containment Spray as a Fission Product Cleanup System," dated March 2007 (ADAMS Accession No. ML070190178), and guidance in SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000 (ADAMS Accession No. ML003734190), are used in the revised LOCA radiological analysis.

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.

Based on the NRC staff's review, the NRC staff has concerns regarding the proposed modeling

of credit for reduction of airborne radioactivity from containment sprays and assumptions regarding reduction of radioactivity in the MSIV leakage pathway presented in the LAR. Resolution of these concerns is needed to complete a technical review and to determine whether compliance with the NRC regulatory requirements in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC 19, and conformance to RG 1.183 and NUREG-0800 (SRP) Sections 6.5.2 and 15.0.1 are met. Therefore, the NRC staff requests the following additional information.

Regulatory basis and background for ARCB-RAI-1A, B & C:

RG 1.183 Appendix A RP 3.3 states that, "Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited." The licensee based the credited spray removal on a spray pump volumetric flow rate of 5,600 gallons per minute (gpm). The licensee assumed that the spray would be initiated by manual action 20 minutes post-accident with an assumed termination at 4 hours and a fall height of 31 feet based on the difference in elevations between the lower spray head and the drywell floor.

The NRC staff examined the JAFNPP UFSAR for evidence that the containment spray systems were designed to provide a reduction in airborne activity in accordance with SRP 6.5.2. Based on this examination it appears that the spray system was designed for pressure reduction and not specially for reducing airborne radioactivity. The NRC staff notes that containment spray design requirements regarding the ability to reduce airborne radioactivity are discussed in Calculation No. JAF-CALC-19-00005 Rev. 0, Section 12.3 NUREG-0800 Section 6.5.2 review items.

The NRC staff examined the calculation of the particulate removal coefficient as documented in Calculation No. JAF-CALC-19-00005 Rev. 0, page 26. Based on this examination it appears that a spray drop fall height of 31 feet was determined by subtracting the elevation of the drywell floor from the elevation of the lower spray header. This method would be valid if there were no obstructions present in the drywell however this is clearly not the case in a Mark I drywell. In addition, the analysis assumes a spray flow rate of 5,600 gpm. Assuming the full spray flow rate of 5,600 gal/minute would be valid if there were no obstructions present in the drywell however this is clearly not the case in a Mark I drywell. The NRC staff notes that it is the unobstructed free fall height that is of importance in the determination of the ability of the containment spray to effectively reduce airborne radioactivity. These issues related to reductions in fall height and spray flow rate resulting from impingement were addressed in the application submitted previously by another licensee for implementation of the AST.

NUREG/CR-5966, A Simplified Model of Aerosol Removal by Containment Sprays, Section H, (ADAMS Accession No. ML063480542) discusses the issue of obstructions interfering with the effectiveness of sprays as follows:

> H. Droplet-Structure Interactions
>
> Reactor containment buildings are not simple, open volumes. Immediately below spray headers there is often a substantial open space. But, eventually, falling drops begin to encounter equipment, structures and operating floor of the reactor. The drywells of Mark I containments are well-known for the congestion that can interfere in the free fall of water droplets.

> The flooring in many reactor containments is grating or so-called "expanded sheet metal." Below the flooring are large volumes which, in a severe reactor accident, would hold aerosol-contaminated gas. It is of interest to know, then, if spray droplets, after hitting structures and the open flooring, would continue to sweep aerosols from the containment atmosphere. Certainly, in the case of the design basis analysis of iodine removal from containment atmospheres, it has been traditional to assume droplets are ineffective once they have hit a structure or the flooring.

ARCB-RAI-1A

Please provide additional information describing how the design characteristics of the containment spray systems regarding the ability to provide a reduction in airborne activity in accordance with SRP 6.5.2, as discussed in Calculation No. JAF-CALC-19-00005 Rev. 0, will be incorporated into the JAFNPP UFSAR.

ARCB-RAI-1B

Please provide additional information providing a justification for the use of the fall height of 31 feet in the determination of the particulate removal coefficient, which apparently does not consider obstructions present in the drywell that would significantly limit the effective fall height.

ARCB-RAI-1C

Please provide additional information providing a justification for assuming the full spray flow rate of 5,600 gallons per minute in the determination of the particulate removal coefficient, which apparently does not consider obstructions present in the in the drywell that would significantly limit the ability of the spray to remove airborne radioactivity in the drywell atmosphere.

Regulatory basis and background for ARCB-RAI-2:

In Attachment 1, Section 3.11.11, the licensee states:

> The AST 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 6.16), Limerick (Reference 6.17), and LaSalle (Reference 6.18). 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….

Although the staff is not questioning the 20-group methodology, 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 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. Another licensee

addressed the staff's issue previously by reducing the settling velocity and the staff found the reduced value was sufficiently conservative (along with other conservatisms) to reflect the effectiveness of the sprays.

The issue of how the shift in the aerosol size because of drywell sprays would impact assumptions made in the subsequent main steam line aerosol deposition was discussed in the pre-application meeting held on June 20, 2019, (ADAMS Accession No. ML19183A128). As stated in the meeting summary, "Since the precedent cited for the proposed main steam line aerosol deposition did not include drywell sprays, the licensee should consider including a detailed discussion of how the use of sprays is accounted for in the subsequent steam line aerosol deposition."

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.

ARCB-RAI-2:

Please provide additional information describing how the gravitational settling credited in the main steam lines considers the changing aerosol characteristics (i.e., aerosol size and density distributions) due to the preferential removal of larger aerosols because of the credit assigned to containment sprays.

Regulatory basis and background for ARCB-RAI-3:

In Attachment 1, Section 3.11.11, the licensee states:

> The AST 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 6.16), Limerick (Reference 6.17), and LaSalle (Reference 6.18). 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 cited precedents included a ruptured MSL to maximize the dose consequences from MSIV leakage. 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 Section 3.11.1, Recirculation Line Rupture Vs Main Steam Line Rupture, which argues that because the recirculation presents a greater challenge to

selective aspects of facility design, a recirculation line rupture is the limiting event with respect to radiological consequences. 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 § 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 is consistent with the guidance from RG 1.183 that assumptions should be selected with the objective of maximizing the postulated radiological consequences.

The NRC staff notes that Calculation No. JAF-CALC-19-00005 Rev. 0, Section 2.3.4 includes a discussion of the basis for assuming a recirculation line rupture instead of ruptured MSL in the assessment of MSIV leakage stating that:

> Although postulating a main steam line break in one steam line inside the drywell would maximize the dose contribution from the MSIV leakage, the steam line break is not a credible event during a LOCA, because the Seismic Class 1 main steam piping up to the TSVs [turbine stop valves] is designed to withstand the SSE [safe shutdown earthquake] (Ref. 9.62).

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.

ARCB-RAI-3

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.