UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 31, 2015

Mr. Michael P. Gallagher
Vice President, License Renewal Projects
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: LICENSE RENEWAL ENVIRONMENTAL SITE AUDIT REGARDING LASALLE COUNTY STATION, UNITS 1 AND 2 - SEVERE ACCIDENT MITIGATION ALTERNATIVES (TAC NOS. MF5567 AND MF5568)

Dear Mr. Gallagher:

On December 9, 2014, the U.S. Nuclear Regulatory Commission (NRC) received the Exelon Generation Company, LLC, application for renewal of operating licenses NPF-11 and NPF-18 for LaSalle County Station, Units 1 and 2. The staff of the U.S. Nuclear Regulatory Commission (the staff) is reviewing this application.

As a part of the environmental review, the staff plans to conduct a severe accident mitigation alternatives (SAMA) audit at the Exelon office in Kennett Square, PA, during the week of April 6, 2015, in accordance with the enclosed SAMA audit plan. The audit plan was informally provided to your staff, Ms. Nancy Ranek, by e-mail on March 23, 2015. If you have any questions, please contact me at 301-415-6223 or by e-mail at david.drucker@nrc.gov.

Sincerely,

/RA/

David Drucker, Sr. Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosure:
As stated

cc w/encl: Listserv

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Letter to Michael Gallagher from David Drucker dated March 31, 2015

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LaSalle County Station (LSCS), Units 1 and 2

Environmental Site Audit Plan
Severe Accident Mitigating Alternatives (SAMA)

GENERAL

Review of SAMA analysis and results as documented in Exelon's License Renewal Environmental Report and supporting documents.

Review of Level 1, 2 and 3 probabilistic risk assessment (PRA) as described in Exelon's License Renewal Environmental Report (ER) and supporting documents.

See pages 2 – 7 of this enclosure for specific questions to be discussed and/or addressed at the audit. Note that additional questions may be developed and provided several days before the audit or during the audit.

DOCUMENTS REQUESTED TO BE AVAILABLE FOR REVIEW

2008 peer review report

Documentation of the review of the 2013A Level 2 model

Methods for Estimation of Leakages and Consequences of Releases (MELCOR) Accident Consequence Code System, Version 2 (MACCS2) calculation documents

Supporting documentation for the core inventory calculation

Documentation of the calculated 2043 population presented in ER Table F.3-7

INTERVIEWS

NRC staff needs to interview Exelon/contractor personnel knowledgeable of LSCS Level 1, 2 and 3 PRA development and results as well as SAMA identification and evaluation. Interviews are to discuss the attached questions and any other questions that may arise from the material reviewed during the audit.

Because responses to several of the questions below (5.a.ii, 6.d and 6.f) are expected to involve detailed discussions of the LSCS design and associated modeling, consideration can be given to scheduling these discussions with specific NRC staff and/or Exelon personnel.

TOUR

No specific tour needed for SAMA.

LaSalle County Station SAMA Audit Questions

1. Level 1 PRA

   a. Section F.2 of the ER states that the LSCS PRA is a Unit 2 model, but because the units are nearly identical, it is considered to be applicable to Unit 1 unless otherwise noted. Provide a brief description of the differences between the units, particularly those differences that might impact internal flooding and their impacts on the unit risk.

   b. Provide the contributions to internal events Core Damage Frequency (CDF) from Station Blackout (SBO) and Anticipated Transients Without Scram (ATWS) scenarios.

   c. Identify the plant systems shared or that could be cross-tied between units and describe the modeling, including the treatment of unavailability during outages or accidents involving the other unit.

   d. Section F.2.1 of the ER indicates that no significant plant modifications affecting the risk profile were performed since the PRA model freeze date. Other than the hard pipe vent, identify any planned changes or modifications to the LSCS design, operation, reactor power level, fuel cycle, or fuel design that might impact the SAMA analysis and describe their expected impacts.

   e. Table F.2-1 of the ER includes a number of footnote citations, but the content of the footnotes are not included. Provide the footnotes.

   f. Section F.2.2.4 of the ER states that the 2000A model included a seismic PRA model (removed when the 2006A model was developed). Confirm that the seismic risk is not included in the results given in Table F.2-1 of the ER. If included, provide the results without the seismic risk.

   g. Provide a summary of the significant changes incorporated in the 2006A PRA model.

   h. Changes incorporated into the 2011A model included assignment of mitigated ATWS scenarios from Accident Class IV to Accident Class II in Section F.2.2.11 of the ER. However, Table F.2-3 of the ER does not appear to include any Class II ATWS scenarios. Provide more detail on the assignment of ATWS scenarios, including any impacts from the 2013A Level 2 model upgrade.

2. Level 2 PRA

   a. Section F.2.2.11 of the ER is titled “2013A Upgrade.” Describe the changes in the Level 2 model that are considered upgrades as opposed to updates. Also, describe the steps taken to assure technical adequacy of the 2013A Level 2 model.

   b. Provide a description of the interface between the Level 1 CDF analysis and the Level 2 analysis, including the use of plant damage states or core damage accident classes and how the two models are linked.

   c. Provide more information on the containment event trees (CETs) utilized in the Level 2 analysis including: the number of CETs, the sequences handled by each CET, the events included in each CET, how the loss of offsite power (LOOP) and SBO sequences are addressed, and the consideration of the impact of hydrogen leakage from containment resulting from containment seal/penetration degradation.

d. Table F.2-6 of the ER includes Accident Classes II, IIE, IIV and IIVE. Classes IIE and IIVE are not defined in Table F.2-3. Define these accident classes.

e. It is noted in Table F.2-6 of the ER that the majority of Class IV (ATWS) sequences result in a low/early (L/E) release and, as indicated in Table F.3-17 for the majority of these sequences, core melt is arrested in the reactor vessel. Discuss the reasons and basis for these results.

f. In Table F.3-17 of the ER, the reference Modular Accident Analysis Program (MAAP) case for Sequence IV-04, which is the dominant contributor to the low/early (L/E) release category frequency, is stated not to have been used for the release fractions because this MAAP case does not model the core melt arresting in the reactor vessel. Another MAAP case, with a release fraction one sixty-fifth of that for the ATWS case, was used to determine the release fractions for the L/E release category. Provide additional support for the use of these release fractions.

g. The run time for several of the LSCS MAAP cases in Table F.3-18 of the ER is less than 48 hours past the declaration of a general emergency when effective offsite resources might be available for mitigating the accident. While it is stated that the representative MAAP cases were run until plateaus of the cesium iodide (CsI) and cesium hydroxide (CsOH) release fractions were achieved, provide more information to justify the adequacy of the MAAP release fraction results for the SAMA analysis for those cases with run times less than 48 hours after the general emergency.

h. In Table F.3-18, the time of four hours for declaration of a general emergency for the moderate/intermediate (M/I) is indicated to have been determined probabilistically. Describe how this time was determined and the impact on the SAMA cost-benefit analysis if the minimum steam coolant water level limit (MSCWLL) time of 27.1 hours is used in the analysis.

3. External Events

a. Section F.5.1.6.2.1 indicates that the impact of using the LSCS 2013 seismic hazard curves on the Risk Methods Integration and Evaluation Program (RMIEP) analysis has been investigated. Provide more information on the source of these hazard curves including a characterization of how they compare with the LSCS hazard curves included in the industry's response to the NRC's Fukushima Daiichi Nuclear Power Plant Near Term Task Force (NTTF) Recommendation 2.1. Also provide the LSCS seismic CDF using the updated hazard curves and the simplified methodology of generic issue GI-199.

4. Level 3 Consequence Analysis

a. Indicate the Sector Population and Economic Estimator (SECPOP2000) version used in the analysis. Confirm that manual entry of economic values was sufficient to prevent known SECPOP2000 errors, described in NRC Request 4.b on page E1-24 of ML102100588, from influencing the SAMA results.

b. Compared to higher (total) estimates in Table F-3.2 for an assumed uniformly distributed population, provide supporting justification that 2010 population estimates in Table F-3.5 are more representative of the actual 2010 population distribution.

c. Indicate the source of the county population growth rates in Table F.3-1.

d. Compare the more recent 2012 Census of Agriculture data to values used in the analysis from the 2007 Census of Agriculture with any consumer price index updates.

e. Section F.3.7 of the ER mentions greater plume washout and radionuclide deposition during precipitation events. Estimate the sensitivity of the offsite population dose and offsite economic cost risk to atmospheric conditions with and without precipitation events.

f. Confirm that a modeling assumption for artificially inducing precipitation at the outer radial cells was not included in the consequence analysis.

g. Specify the software codes and versions used for calculating the core inventory.

5. Selection and Screening of Phase I SAMAs

a. Relative to the identification of SAMAs from the Level 1 importance review documented in Table F.5-1:

i. The turbine trip with bypass initiating event (%TT) is indicated to have a frequency of 0.8 per year, which implies that a number have occurred over the LSCS operating history. Do any of these occurrences suggest possible cost-beneficial SAMAs?

ii. The discussion of potential SAMAs for a significant number of events (RCVCL-1BE, %TIA, RCVSEQ-DLOP-041, BWTOPWTHXSTBYH--, and others) involve water hammer induced loss of coolant accidents (LOCAs) related to the generation of a LOCA signal on high drywell pressures when an actual LOCA does not exist and are mitigated by SAMA 7. Provide more detail on the water hammer scenarios in general and more specifically those involving the above identified events including: how loss of instrument air initiated events involve water hammer induced LOCAs, why alignment of the standby TBCCW heat exchanger train is needed, why water hammer LOCAs are not important for other non-LOCA transients, and the potential for preventing water hammer by preventing residual heat removal (RHR) pump start when water hammer conditions exist or preventing draining of the RHR line. Also, clarify the SAMA 7 statement in Table F.5-4 of the ER that the LOOP-delayed LOCA scenario is not specifically modeled in the PRA.

iii. For event 2ADRX-TRANS--H-- (p. F-244 of the ER), failure to manually depressurize the reactor pressure vessel, the action is indicated to be required for about 90% of the scenarios because failure to initiate suppression pool cooling (SPC) results in containment failure. Discuss these scenarios in more detail including the reason for this high percentage compared to other causes of failure of high pressure injection.

iv. For event 2VYFNNWVY01--X-- (p. F-271 of the ER), VY NW corner room (RHR A) cooling fan 2VY01C fails to run, previous LSCS evaluations could not demonstrate that portable fans would provide adequate cooling for the reactor building corner rooms when the normal cooling system failed. Discuss this previous evaluation and if sufficient cooling can be achieved if the portable fans can provide air movement through the room coolers assuming cooling water remains available.

b. Relative to the identification of SAMA from the Level 2 importance review documented in Table F.5-2:

   i. For event 2TDOP-RECLPS2H-- (p. F-291 of the ER), Operator fails to recover low-pressure systems, the discussion indicates that low-pressure systems would not have the power to function. Consider the potential for utilizing a fire truck or other portable self-powered pumps for injection into the containment.

   ii. Event 2GVPHCMBSTGASF-- (p. F-286, F-300, and F-316), Combustible gas venting fails, appears in all three Level 2 importance lists. Discuss this event and if any SAMAs are possible to directly mitigate this failure.

c. The discussion in Section F.5.1.3.6 of Grand Gulf SAMA 59 states that “Rather than cycling large pumps in scenarios where the cooling system is lost, a more effective means of maintaining injection with the Emergency Core Cooling System (ECCS) pumps is considered to be through the use of portable/temporary cooling alignment, which is addressed in the LSCS importance list review by SAMA 16." Evaluate a SAMA similar to Grand Gulf's SAMA 59, Increase operator training for alternating operation of the low pressure emergency core cooling system pumps (low-pressure coolant injection and low pressure core spray) for loss of standby service water scenarios, for rooms where the use of portable fans may not be effective.

d. While the Commonwealth Edison Individual Plant Examination (IPE) submittal did not identify any vulnerability, as discussed in the NRC’s IPE Safety Evaluation Report, the IPE did cite NUREG/CR-4832, Vol. 3, Part 1 for insights about potential improvements. These are: (1) Eliminate the sneak circuit in the reactor core isolation cooling (RCIC) isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite alternating current (AC) power is lost and the appropriate diesel generator starts, (2) Change the RCIC room temperature isolation logic so that, in cases where AC power from train A has failed but AC power from train B is available, this isolation logic does not isolate if no other emergency core cooling system is working, and (3) Change the venting procedure so that venting does not result in severe environments in the reactor building. Discuss the current status of these potential improvements.

e. The RMIEP Summary (NUREG/CR-4832, Volume 1) identifies the common cause failures of the core standby cooling system (CSCS) cooling water pumps as the dominant events in the seismic risk reduction importance assessment. Discuss these events and their importance in the seismic analysis used for SAMA identification.

f. Relative to the identification of SAMAs to mitigate internal fires:

   i. As discussed in Section F.5.1.6.1 of the ER, the lower cutoff value of 5% of the total fire CDF is used for considering SAMAs. Using this cutoff value results in not identifying potential cost-beneficial SAMAs below a cost of $142,000. Evaluate additional fire sequences below this threshold in order to identify potential low cost procedure enhancement SAMAs.

ii. As indicated in Section F.5.1.6.1.1 of the ER, fires in the Division 1 and Division 2 essential switchgear rooms for each unit make up 60% of the total fire CDF. The only SAMA proposed for mitigating these fires is SAMA 1, installing a reliable hard pipe vent. SAMA 1 only mitigates the adverse consequences of venting and does not mitigate the direct impact of the fire. Discuss these fire scenarios and the potential for other SAMAs to directly mitigate the fire at an earlier stage of the scenario or by mitigating events in the cutsets other than the adverse conditions due to venting.

iii. Section F.5.1.6.1.3 of the ER indicates that the largest contributing fire scenario to the Auxiliary Electrical Equipment Room fire risk is a bounding cable fire caused by hot work. Discuss the potential for a SAMA to mitigate this risk.

iv. Section F.5.1.6.1.3 of the ER also indicates that several individual panel fires are also key contributors to the overall Auxiliary Electrical Equipment Room fire risk profile. The largest contributing scenarios (M, B, and C for Unit 1; M, E, and J for Unit 2) are stated to contribute 80% of the total fire zone frequency. This appears inconsistent with the prior statement that the largest contributing fire scenario to the Auxiliary Electrical Equipment Room fire risk is a bounding cable fire caused by hot work. Provide more information on the panel fire risk contributions to fire risk and discuss the potential for SAMAs to directly mitigate this risk contribution.

v. The RMIEP Summary (NUREG/CR-4832, Volume 1) identifies the three most important events in the fire analysis risk reduction importance assessment to be in the control room abandonment scenario. Discuss this scenario and its importance in the current fire analysis and the potential for a cost-beneficial SAMA to mitigate this scenario.

6. Cost-Benefit Analysis

a. Clarify if maintenance costs are included in the estimated implementation costs.

b. Confirm that the assumptions concerning the design of SAMA 1 are consistent with NRC Order Number EA-13-109, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions."

c. Discussions of SAMA 8 in Section F.6.8 of the ER and SAMA 14 in Section F.6.13 of the ER include the statement "Flow from the fire protection system, in its current configuration, is only adequate in cases where RCIC....". Discuss what is meant by "current configuration" and the potential for a SAMA to address the fire protection system configuration and make its possible use without prior RCIC operation.

d. Discussion of SAMA 14 (Provide a portable DC source to support RCIC and SRV operation) in Section F.6.13 of the ER indicates that the PRA model was changed to include a lumped event to represent the 480V AC power source that feeds the division 1 battery chargers. The SAMA description and the basic events cited to be mitigated by this SAMA indicated that a direct current (DC) power source to directly supply an ESF DC distribution panel is needed. Discuss the inclusion of an AC power supply in the model.

e. SAMA 7 (Water hammer prevention) and SAMA 25 (Periodic training on water hammer scenarios resulting from a false LOCA signal) both are intended to mitigate the water hammer scenarios involving SPC operation interrupted by a LOOP. The changes made to the model and the impact of the changes on person-rem and offsite economic cost risk are significantly different for the two SAMAs. Discuss these SAMAs and their analyses in more detail to justify these differences.

f. A few SAMAs (e.g. 8, 14 and 27) involve use of equipment that may be available as a result of the B.5.b Program. Discuss this further and the impact on the cost-benefit analysis.

g. SAMA 9 (Develop flood zone specific procedures) and SAMA 11 (Provide the capability to trip the FPS pumps) both address internal flooding whose principal contributor is a fire protection system (FPS) pipe rupture in the reactor building. The assumption for evaluating SAMA 9, that the risk from all internal floods will be eliminated, results in a 9% reduction in CDF and this SAMA being cost-beneficial. The assessment of SAMA 14 results in less than a 2% reduction in CDF; while as stated in Section F.6.11, SAMA 11 is designed to eliminate the FPS's flow. Discuss the FPS design and the FPS pipe break scenarios and associated modeling to support the above results.

h. Confirm that SAMA 20 for improving the vacuum breaker reliability is not related to managing the effects of plant aging during the period of extended operation.

7. Lower Cost Alternative SAMAs

   a. Consider the potential for changes to the suppression pool cooling operating procedures/practices to reduce the chance of the 2 psi high drywell signal being reached for normal transients as an alternative to SAMA 7.