Exelon Generation®

200 Exelon Way
Kennett Square, PA 19348
www.exeloncorp.com

10 CFR 50.90
10 CFR 50.69

August 14, 2017

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

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Supplement to Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors

References:

1. Letter from J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants'," dated June 28, 2017 (ADAMS Accession No. ML17179A161).

2. Letter from V. Sreenivas (U.S. Nuclear Regulatory Commission) to B. C. Hanson (Exelon Generation Company, LLC), "Limerick Generating Station, Units 1 and 2, – Supplemental Information Needed for Acceptance of Requested Licensing Action RE: Adoption of Title 10 of the Code of Federal Regulations Section 50.69 (CAC Nos. MF9873 and MF9874)," dated July 31, 2017 (ADAMS Accession No. ML17207A077).

In Reference 1, Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (Limerick), Units 1 and 2, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors.

In Reference 2, the NRC requested that Exelon provide supplemental information by August 17, 2017 to support the acceptance review of the license amendment request. The attachment to this letter provides a restatement of the NRC questions followed by our responses.

Exelon has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in the Enclosure of the Reference 1 letter. Exelon has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, Exelon has concluded that the information in this response does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this supplement to the application for license amendment by transmitting a copy of this letter and its attachment to the designated State Official.

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 14th day of August 2017.

Respectfully,

James Barstow
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachment

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, Limerick
USNRC Senior Resident Inspector, Limerick
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection

ATTACHMENT

License Amendment Request Supplement

Limerick Generating Station, Units 1 and 2
NRC Docket Nos. 50-352 and 50-353

Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors

In Reference 1, Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (Limerick), Units 1 and 2, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors.

In Reference 2, the NRC requested that Exelon provide supplemental information by August 17, 2017 to support the acceptance review of the license amendment request. A restatement of the NRC questions followed by our responses is provided below.

1. The regulations in 10 CFR 50.69(c)(1)(i) require that the probabilistic risk assessment (PRA) must be (1) of sufficient quality and level of detail to support the categorization process and must be (2) subjected to a peer review process assessed against a standard or set of acceptance criteria endorsed by the NRC. Section 50.69(b)(2)(iii) of 10 CFR requires that the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) criteria be submitted as part of the application. Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides guidance for determining the technical adequacy of the PRA by reviewing it against relevant parts of the ASME/ANS Standard RA-Sa-2009 using a peer review process.

   While the NRC staff found the information provided in the 50.69 LAR referenced above, regarding the internal events PRA quality, to be insufficient for detailed technical review, the staff noted that the licensee submitted PRA quality information in a relief request dated April 13, 2016 (ADAMS Accession No. ML16104A122), as supplemented on September 19, 2016 (ADAMS Accession No. ML16263A218), in response to the NRC's request for additional information. In the licensee's submittal pertaining to this relief request, the licensee stated that the 2005 peer review of the internal events PRA was a full-scope peer review against RG 1.200, Revision 0, and provided results of gap assessments to RG 1.200, Revision 2. An overview of all changes to the internal events PRA performed after the 2005 peer review was also provided. This information was used to support the review of the internal events for this 50.69 LAR.

   To support an effective licensing review and reduce unnecessary delays in the review, provide the following information:

   a. The LAR states that a peer review of the internal flooding PRA was performed in 2008 against RG 1.200, Revision 1, and that gap assessments to RG 1.200, Revision 2, were conducted, but no information on these gap assessments were provided in the relief request. To support the LAR statement that the internal flooding PRA model meets the requirements of RG 1.200, Revision 2, provide the gap assessment of the internal flooding PRA against RG 1.200, Revision 2.

      Response

      As indicated in the NRC question, results of the gap assessment to Revision 2 of RG 1.200 were provided for the internal events PRA in the response to the request for additional information for the RI-ISI submittal (ADAMS Accession No. ML16263A218), but the gap assessment results for the internal flooding PRA were not provided in that response since there was no impact from internal flood hazards on the RI-ISI analysis employed for Limerick. The results of the gap assessment for the Internal Flooding (IF) Supporting Requirements (SRs) identified in NEI 05-04, Revision 3 (Reference 3), are provided below.

Supporting Requirements Requiring Re-evaluation

SRs that require re-evaluation are those SRs that have changed significantly, including those with new issues identified in RG 1.200, Revision 2. The applicable IF SRs are identified in NEI 05-04, Revision 3 and their impact for Limerick and this application are provided in Table 1.

Table 1: IF SRs Requiring Gap Assessment Evaluation

Supporting Requirement	Comments from NEI 05-04, Revision 3	Impact on Limerick for this 50.69 Application
Flooding SRs: IFPP-B1, B2, B3, IFSO-B1, B2, B3, IFSN-B1, B2, B3, IFEV-B1, B2, B3, IFQU-B1, B2, B3	These are new requirements for flooding that expand on the original SRs in the ASME/ANS PRA Standard.	No impact. Limerick meets the current Capability Category I/II/III requirements for these SRs. The Limerick Internal Flood Evaluation Summary Notebook (LG-PRA-012) (Reference 4) provides the necessary documentation that facilitates PRA applications, upgrades, and peer reviews requirements for each of the IF*-B1 SRs. The Limerick Internal Flood Evaluation Summary Notebook also provides the necessary documentation to meet each of the IF*-B2 SRs. The sources of model uncertainty and related assumptions are documented in Appendix A of the Limerick Internal Events Summary Notebook and are based on the guidance provided in EPRI TR-1016737 (Reference 5), as endorsed in NUREG-1855 (Reference 6). This includes sources of flooding uncertainty. Additionally, the Limerick Internal Flood Evaluation Summary Notebook was updated to include uncertainty and assumptions. Section 2.2 includes

Table 1: IF SRs Requiring Gap Assessment Evaluation		
Supporting Requirement	**Comments from NEI 05-04, Revision 3**	**Impact on Limerick for this 50.69 Application**
		assumptions and Appendix G includes uncertainty and sensitivity evaluations. This information meets the intent of the IF*-B3 SRs.
IFSN-A6	RG 1.200, Revision 2, provides clarification that should be evaluated.	No impact. Now Met Capability Category II per RG 1.200 clarification. As a part of the 2013 FPIE PRA Update, pipe whip effects were investigated and shown to not be a concern for piping containing moderate energy water sources. Jet impingement effects were also shown to not be a concern for piping encapsulated by aluminum lagging. Although the explicit consideration of the other failure mechanisms might ultimately introduce a few additional scenarios, the approach which initially utilizes bounding assumptions regarding the failure of all equipment in the flood area for the initial CCDP determination would bound the potential risk increase associated with these low likelihood events. This is sufficient for meeting Capability Category II including the RG 1.200 clarification.

In summary, a gap assessment to the current standard, ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 has been performed. The gap assessment did not identify any deficiencies that were not identified by the peer reviews or were not previously self-identified with respect to the new standard, and the remaining open items are consistent with the 2016 independent review team conclusions. The results of the technical adequacy evaluation (including internal flooding) and their impact on this application were provided in Attachment 3 of the Limerick 50.69 LAR submittal (ADAMS Accession Number ML17179A161).

b. Confirm that the peer review conducted in 2011 for the fire PRA was a full-scope peer review and followed Nuclear Energy Institute (NEI) 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines." If the review was not a full-scope peer review, please describe the review in detail and provide all earlier findings and observations from any previous peer reviews.

<u>Response</u>

The Limerick Fire PRA peer review conducted in 2011 was a full scope peer review in accordance with the guidance in NEI 07-12, Revision 1 (Reference 7).

c. Confirm that the fire PRA uses methods that have been formally accepted by the NRC staff. If there are any methods used in the fire PRA that have not been formally accepted, describe the method and provide adequate technical justification for the method.

<u>Response</u>

The Limerick Fire PRA uses methods that have been formally accepted by the NRC.

2. The guidance in Section 5 of NEI 00-04, "10 CFR 50.69, SSC Categorization Guideline," as endorsed by RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," stipulates identification of any applicable sensitivity studies to be used during the categorization process that are associated with the licensee's choice of specific models and assumptions, as discussed in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The LAR states that PRA model-specific assumptions and sources of uncertainty for this application have been identified and dispositioned but did not provide a description of the evaluated uncertainties and their disposition.

   Provide the technical justification to support the LAR conclusion that no additional sensitivity analyses are required for the categorization process.

   <u>Response</u>

   The baseline internal events PRA and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the 50.69 Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the 50.69 Program.

   The epistemic uncertainty analysis approach described below applies to the internal events PRA. Epistemic uncertainty impacts that are unique to FPRA are addressed following the internal events discussion.

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for 50.69 Program application, the internal events baseline PRA model uncertainty report, which was developed based on the guidance in NUREG-1855 and EPRI TR-1016737, was reviewed. As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Based on following the methodology in EPRI TR-1016737 for a review of sources of uncertainty, the impact of potential sources of uncertainty on the 50.69 application is discussed in Table 2, which identifies those sources that have the potential to be key sources of uncertainty for the 50.69 program.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
The Loss of Offsite Power (LOOP) frequency and fail to recover offsite power probabilities are based on available industry data.	SSCs that support LOOP scenarios	The overall approach for the LOOP frequency and fail to recover probabilities utilized is consistent with industry practice and are representative of Limerick. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
Recovery of instrument air (IA) is assumed to be possible to support containment venting in loss of containment heat removal scenarios.	SSCs that support containment heat removal scenarios	Given the diversity and redundancy of the IA systems at the site, credit for IA recovery (e.g., by aligning to the opposite unit compressors) for success of containment venting in long term loss of decay heat scenarios is reasonable. A slight conservative bias slant is used for this recovery value such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Continued injection from control rod drive (CRD) after containment failure is credited unless a gross rupture of containment (i.e., not leak before break) occurs. The probability of rupture is based on a detailed structural analysis of the Mark II design.	SSCs that support containment heat removal scenarios	This approach provides a best estimate assessment for the site. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
The base PRA model includes an assumption that 2 emergency diesel generator (EDG) HVAC fans are required 25% of the time, and only 1 EDG HVAC fan is required for the remaining 75% of the time.	SSCs supporting scenarios in which on-site AC power is required	This approach provides a best estimate assessment for the site. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
The base PRA model credits serial operation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) to provide initial injection out to four hours in LOOP and Station Blackout (SBO) scenarios without explicit representation of load shedding.	SSCs supporting scenarios in which on-site AC power is required	Prior to implementation of the 50.69 program, the PRA model will be updated to explicitly account for load shedding when procedurally directed. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
The postulated reactor pressure vessel (RPV) overpressure failure mode is assumed to be equivalent to the Large LOCA success criteria.	SSCs supporting the LPI function in RPV overpressure failure LOCA scenarios	An alternative assumption would be that such scenarios are beyond the capabilities of the LPI systems. Therefore, crediting LPI capabilities for these scenarios may provide a slight non-conservative bias on the 50.69 calculations. However, because RPV overpressure LOCA scenarios are very low frequency events, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
The pipe rupture frequencies in the internal flooding PRA are based on an older version of the EPRI pipe rupture frequencies. Conversion to the most recent EPRI pipe rupture frequencies may increase internal flood CDF. The internal flood model uses a pipe length approach per EPRI TR-1013141 (Reference 8). Newer data is available.	SSCs that support Internal Flood scenarios	Prior to implementation of the 50.69 program, the internal flood model will be updated so that the model uses the newer frequencies. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
Credit for core melt arrest in-vessel at high RPV pressure conditions is taken in the current PRA model, but with a nominal failure probability of 0.9.	SSCs that support LERF scenarios	Core melt arrest in-vessel at high pressure may not be possible and therefore this could be a source of model uncertainty. Use of the 0.9 factor compared to the alternative assumption of 1.0 would not have a meaningful impact on the 50.69 calculations. However, prior to implementation of the 50.69 program, the PRA model will be updated to change this value to 1.0, such that this does not represent a key source of uncertainty for the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Timely low pressure emergency core cooling system (ECCS) restoration after core damage is assumed to lead to a condition where vessel failure is avoided.	SSCs that support LERF scenarios	This assumption precludes some of the low likelihood phenomenological contributors to LERF from contributing to the overall results. However, it is judged reasonable that the availability of low pressure injection at the time of vessel failure (should it occur) will also greatly reduce the potential for a large early release from occurring. Therefore, this assumption provides a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
If containment failure occurs prior to core damage in Anticipated Transient Without Scram (ATWS) scenarios that could result in LERF, only injection from residual heat removal service water (RHRSW) is credited to provide core melt arrest in-vessel. Besides the failure modes of implementing RHRSW injection, additional failure modes are included for harsh reactor building environment or piping failures due to containment failure.	SSCs supporting ATWS LERF scenarios	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
Ex-vessel core melt progression overwhelming vapor suppression is considered in the LERF model with different values for low pressure RPV failure sequences and high pressure RPV failure sequences based on available information.	SSCs that support LERF scenarios	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
ISLOCAs are dominant contributors to LERF. Their assumed IE frequency could influence the LERF FV and RAW values of all SSCs. The detailed Interfacing System LOCA (ISLOCA) analysis includes the relevant considerations listed in IE-C14 of ASME/ANS PRA Standard RA-Sa-2009 (Reference 9) and accounts for common cause failures and captures likelihood of different piping failure modes.	SSCs that support LERF scenarios	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
Given that conditions occur that would allow uncontrolled flooding of the steam lines, a probability is assigned that this uncontrolled flooding permanently disables all of the SRVs precluding the ability to depressurize the RPV through the SRVs.	SSCs that support scenarios that require High Pressure Injection	Although the SRVs at Limerick are designed to pass water and Appendix R models the RPV being flooded with water returned to the Suppression Pool via the SRVs, they are never tested in this fashion. A nominal failure probability is assigned to provide a slight conservative bias slant to the results such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application
EDG repair probabilities employed in the PRA model are a potential source of uncertainty.	SSCs supporting scenarios in which on-site AC power is required	No credit for EDG repair is taken in the current PRA model. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
Residual heat removal (RHR), RHRSW, and emergency service water (ESW) pump repair probabilities are a potential source of uncertainty.	SSCs that support containment heat removal scenarios	No credit for RHR, RHRSW, or ESW pump repair is taken in the current PRA model. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Containment integrity following a vessel rupture event (i.e., excessive LOCA) is not assured. There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event.	SSCs that support LERF scenarios	The current model treatment results in addition of a constant adder to the CDF and LERF results and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
Digital feedwater control failure probabilities are derived from the reliability values in the vendor study (LG-PRA-005.04) (Reference 10) demonstrating that the system performance would result in less than 0.1 transients per year and these reliability values are used for the key components of the system.	SSCs that support scenarios that require High Pressure Injection	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Uncertainties associated with the assumptions and method of calculation of Human Error Probabilities (HEPs) for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all SSCs evaluated during 50.69 categorization	Sensitivity cases performed using the base internal events PRA (HEP values of 0.0 or use of the 95th percentile value HEPs) indicate some sensitivity to human performance. Use of 95th percentile HEPs for applications is not considered realistic given the consistent use of a consensus HRA approach. The Limerick PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. However, as directed by the guidance to the 50.69 process, the 0 and 95th percentile values of the PRA HEPs are evaluated in the 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Dependent HEP values are developed for significant combinations of HEPs that have been demonstrated to appear together in the same cutsets.	Potentially all SSCs evaluated during 50.69 categorization	The Limerick PRA model is based on industry consensus modeling approaches for its dependent HEP identification and calculations, so this is not considered a significant source of epistemic uncertainty. However, as directed by the guidance to the 50.69 process, the 0 and 95th percentile values of the PRA dependent HEPs are used in the 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

Table 2: Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Common cause failure values are developed using available industry data.	Potentially all SSCs evaluated during 50.69 categorization	The Limerick PRA model is based on industry consensus modeling approaches for its common cause identification and value determination, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations. Additionally, as directed by the guidance to the 50.69 process, the 5^{th} and 95^{th} percentile values of the PRA CCF alpha factors are evaluated in the 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled CCF probabilities are accounted for in the 50.69 application.
There are model uncertainties associated with modeling the probability of the RHR pumps failing from a rupture due to a water hammer event given the RHR system is operating in suppression pool cooling mode at the time of the initiating event and the appropriate operator responses do not occur such that a potential water hammer event can occur.	SSCs supporting scenarios requiring RHR or RHRSW systems	The water hammer basic events and values utilized provide a reasonable best-estimate approach and will have only a minor impact on the 50.69 calculations. This does not represent a key source of uncertainty for the 50.69 application.

Assessment of Fire PRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the Limerick FPRA. The Limerick FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The development of the Limerick FPRA was guided by NUREG/CR-6850 (Reference 11). The Limerick FPRA model used consensus models described in NUREG/CR-6850.

Limerick used guidance provided in NUREG/CR-6850 and NUREG-1855 to address uncertainties associated with FPRA for the 50.69 application. As stated in Section 1.5 of NUREG-1855:

> "Although the guidance does not currently address all sources of uncertainty, the guidance provided on the process for their identification and characterization and for how to factor the results into the decision making is generic and is independent of the specific source. Consequently, the process is applicable for other sources such as internal fire, external events, and low power and shutdown."

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> "A source of model uncertainty is one that is related to an issue in which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion and introduction of a new initiating event)."

NUREG-1855 defines consensus model as:

> "A model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."

The potential sources of model uncertainty in the Limerick FPRA model were characterized for the 16 tasks identified by NUREG/CR-6850 in Table 3. This framework was used to organize the assessment of baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on 50.69 calculations. Table 3 outlines sources of uncertainties by task and their disposition.

As noted above, the Limerick FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Further, appropriate cable impacts were identified for the systems modeled in the Internal Events PRA and were modeled in the Fire PRA. No systems were conservatively assumed to be failed for all FPRA scenarios. Fire PRA methods were based on NUREG/CR-6850, other more recent NUREGs (e.g., NUREG-7150), and published "frequently asked questions" (FAQs) for the FPRA.

In addition to the discussion of sources of model uncertainty in Table 3, the evaluation of sources of model uncertainty in the FPRA and associated sensitivity studies identified one modeling uncertainty that may be potentially significant for applications. See Table 4 for details.

Table 3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the BWROG Generic MSO list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
4	Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
5	Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development as was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. Limerick uses the ignition frequencies in NUREG-2169 (Reference 12) along with the revised heat release rates from NUREG 2178 (Reference 13).	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Consensus approaches are employed in the model. Therefore, the 50.69 calculations are not impacted.
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The Limerick FPRA did not screen out any fire scenarios based on low CDF/LERF contribution. That is, quantified fire scenarios results are retained in the cumulative CDF/LERF. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG 7150, Volume 2 (Reference 14), based on actual fire test data, were used in the Limerick Fire PRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG 7150, Volume 2. The uncertainty values specified in NUREG 7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG 7150, Volume 2. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
11	Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty	Consensus modeling approach is used for the Detailed Fire Modeling. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
		arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	
12	Post-Fire Human Reliability Analysis	The human error probabilities used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The human error probabilities were obtained using the EPRI HRA Calculator (Reference 15) and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The human error probabilities were obtained using the EPRI HRA calculator and included the consideration of degradation or loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. Except as noted in Table 4, it is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	The methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
16	FPRA Documentation	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements.	The methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Table 4: Treatment of Specific Fire PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	50.69 Impact	Model Sensitivity and Disposition
Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all SSCs evaluated during 50.69 categorization.	The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The Limerick FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. However, as directed by the guidance to the 50.69 process, the 0 and 95th percentile values of the PRA independent and dependent HEPs are evaluated in the 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

REFERENCES

1. Letter from J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “Application to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants’," dated June 28, 2017 (ADAMS Accession No. ML17179A161).

2. Letter from V. Sreenivas (U.S. Nuclear Regulatory Commission) to B. C. Hanson (Exelon Generation Company, LLC), "Limerick Generating Station, Units 1 and 2, – Supplemental Information Needed for Acceptance of Requested Licensing Action RE: Adoption of Title 10 of the Code of Federal Regulations Section 50.69 (CAC Nos. MF9873 and MF9874)," dated July 31, 2017 (ADAMS Accession No. ML17207A077).

3. NEI 05-04, Revision 3, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard, November 2009.

4. LG-PRA-012, Revision 2, "Limerick Generating Station Internal Flood Evaluation Summary Notebook,” January 2014.

5. EPRI TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments,” Electric Power Research Institute, Final Report, December 2008.

6. NUREG-1855, “Guidance on the treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” Main Report, March 2009.

7. NEI 07-12, “Fire Probabilistic Risk Assessment Peer Review Process Guidelines,” Rev. 1, June 2010.

8. EPRI TR-1013141, “Pipe Rupture Frequencies for Internal Flooding PRAs,” Rev. 1, Electric Power Research Institute, Palo Alto, CA, 2006.

9. ASME/ANS RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to RAS-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

10. LG-PRA-005.04, Revision 2, “Limerick Generating Station Condensate and Feedwater (COND/FW) System Notebook, Appendix E, January 2014.

11. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.

12. “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009,” NUREG-2169/EPRI 3002002936, U.S. NRC and Electric Power Research Institute, January 2015.

13. "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178 Vol. 1/ EPRI 3002005578, U.S. NRC and Electric Power Research Institute, Draft Report for Comment, April 2015.

14. "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," Final Report, NUREG/CR-7150, Vol. 1, EPRI 3002001989, U.S. NRC and Electric Power Research Institute, May 2014.

15. EPRI HRA Calculator® Software User's Manual, Version 4.21, EPRI, Palo Alto, CA, and Scientech, a Curtiss-Wright Flow Control Company, Tukwila, WA: 2011. Software Product ID #: 1022814.