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U. S. Nuclear Regulatory Commission
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Response to NRC Request for Additional Information Concerning Application for License Amendment to Implement WCAP-14333 and WCAP-15376

References:

1. Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Application for License Amendment: Implementation of WCAP-14333 and WCAP-15376, Reactor Trip System Instrumentation and Engineered Safety Feature Actuation System Instrumentation Test Times, Completion Times and Surveillance Test Intervals," dated January 7, 2007

2. Letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Units 1 and 2 - Request for Additional Information Related to License Amendment Request to Revise Technical Specification Requirements for Select Reactor Trip System, Engineered Safety Feature Actuation System, and Containment Ventilation Isolation, Instrumentation (TAC Nos. MD4009, MD4010, MD4011, and MD4012)," dated August 17, 2007

In Reference 1, Exelon Generation Company, LLC (EGC) submitted a request to amend Appendix A, Technical Specifications (TS) for Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment would revise the TS requirements for selected Reactor Trip System (RTS) instrumentation, Engineered Safety Feature Actuation System (ESFAS) instrumentation, and Containment Ventilation Isolation Instrumentation to adopt Completion

Time, test bypass time, and Surveillance Test Interval changes. The proposed changes are based on Westinghouse Electric Company LLC topical reports WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times."

In Reference 2, the NRC provided a request for additional information (RAI) related to the proposed license amendment. In response to this request, EGC is providing the information in Attachment 1 to this letter.

There is one new regulatory commitment and two revised regulatory commitments described in this response. The new and the revised regulatory commitments are described in Attachment 2. Any other statements in this submittal are provided for information purposes and are not regulatory commitments. Should you have any questions related to this letter, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 12th day of October 2007.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager - Licensing

Attachments:

1. Response to Request for Additional Information, License Amendment Request to Implement WCAP-14333 and WCAP-15376
2. Summary of Regulatory Commitments

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station

NRC Request 1

The analysis for WCAP-14333/WCAP-15376 assumed that maintenance on master and slave relays, logic cabinets, and analog channels while at power occurs only after a component failure, and that preventive maintenance does not occur. The topical report does not preclude the practice of at-power preventive maintenance but limits the total time a component is unavailable due to corrective or preventive maintenance to the values used in the analysis. If preventive maintenance is to be performed at Byron/Braidwood Stations Units 1 and 2, confirm that the unavailability for components evaluated in WCAP-14333 are consistent with the plant specific estimates at Byron/Braidwood and do not exceed those assumed in the analysis.

See the submittal, Attachment 6, Page 1, "Implementation Guidelines," Table 1, "Analog Channel Calibration" as an example.

Response

Exelon Generation Company, LLC (EGC) does not perform at-power preventative maintenance on master and slave relays, logic cabinets, and analog channels.

NRC Request 2

Provide an assessment of external events risk impact including, seismic, fire, and external floods/high wind with respect to the proposed completion times (CT) and surveillance test interval (STI) extensions, per Regulatory Guide (RG) 1.174 Section 2.2.4, "Acceptance Guidelines" and RG 1.177 Section 2.3.2, "Scope of the Probabilistic Risk Assessment (PRA) for technical specification (TS) Applications." Include any seismic vulnerabilities associated with instrumentation/logic systems or components and risk impacts on fire screening criteria and quantification of fire sequences in unscreened areas. In addition, confirm that the combined total CDF from external and internal events remains less than the RG 1.174 base CDF of 1E-4 per year.

Response

In the Braidwood Station and Byron Station Individual Plant Examination of External Events (IPEEE), the dominant risk contribution was from internal fire initiating events. The fire risk quantification approach that was used involved the application of fire-initiating events and fire scenario-related effects on mitigating systems to pertinent internal events in at-power PRA accident sequences. The fire damage impact is typically assigned a probability of 1.0. Thus, any mitigating equipment that might be affected by the postulated fire is assumed to fail in the applicable internal events model scenario (i.e., those scenarios used to represent the plant response to the fire-initiating event). The fire risk is generally driven by this conservative fire damage assumption. The effect of the proposed change will be associated with the test and maintenance unavailabilities for the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) that are used in the internal events model. Thus, the impact of the proposed changes on the test and maintenance unavailabilities are expected to be a second order effect (i.e., smaller) for fire core damage sequences, relative to the effect in the internal events model. Other potential contributors, such as seismic effects and high winds, were within acceptable limits, as discussed below.

The seismic analyses in the Braidwood Station and Byron Station IPEEEs were based on the seismic margin assessment. No significant seismic concerns were identified, and as such, both

Stations possess significant seismic margin. The IPEEEs did identify some seismic outliers (i.e., control room lighting diffuser panels and electrical cabinet interactions). These seismic outliers were resolved with the implementation of design modifications in 2003. The proposed change to the RTS, ESFAS, and Containment Ventilation Isolation Instrumentation Completion Times (CTs) and Surveillance Test Intervals (STIs) has a negligible effect on the seismic risk profile at both stations.

Evaluation of high winds, external floods, and other external events in the Braidwood Station and Byron Station IPEEEs have been completed in accordance with the requirements of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." These evaluations indicated that both Stations conform to the acceptance criteria defined in NUREG-0800, Revision 1, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," with no potential vulnerabilities. The proposed change to the RTS, ESFAS, and Containment Ventilation Isolation Instrumentation CTs and STIs do not impact these conclusions.

Since EGC does not currently have quantitative external event models or internal fire risk models for either Braidwood Station or Byron Station that are sufficiently developed and peer reviewed to enable EGC to combine the risk contribution from external events with that of internal events, EGC has utilized a common approach to estimate the risk from external events and combine this estimate with the risk from internal events. This approach assumes that the risk from external events is equal to the risk from internal events. Additionally, EGC has assumed that the fractional LERF contribution from the internal events model also provides a reasonable estimate of the LERF impact from external events.

EGC has recently approved a revised Braidwood Station and Byron Station PRA. In part, this new model incorporates a revision to the internal flooding analysis, specifically a revision to the internal flooding Human Reliability Analysis (HRA) to meet the requirements of Regulatory Guide 1.200, Revision 0, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

Utilizing this revised model, in concert with the approach described above for estimating the risk contribution from external events, the combined total CDF from internal and external events is below the 1E-04 threshold in RG 1.174, Revision 1 for both Braidwood Station and Byron Station. These results assume that, in part, each Station has implemented, or will implement enhanced internal flood response procedures prior to implementation of the proposed changes. At the current time, Braidwood Station has implemented these procedures. EGC will implement the enhanced internal flood response procedures at Byron Station prior to implementation of the proposed changes. **[Regulatory Commitment]**

<u>NRC Request 3</u>
Provide a discussion on the following aspects of probability risk assessment (PRA) quality for Byron/Braidwood.

<u>NRC Request 3.a</u>
Provide the date of Byron and Braidwood Stations PRA industry peer review and date of certification. Provide the results of the Peer review including disposition of "A" and "B" facts and observations (F&Os).

<u>Response</u>
The Braidwood Station PRA was subjected to a Westinghouse Owners' Group (WOG) peer review in September 1999. The Byron Station PRA was subjected to a separate (WOG) peer review in July 2000. No peer reviews have taken place since that date for either site. Since those peer reviews, the PRA model (the Braidwood Station and Byron Station PRA models are very similar and exist in an integrated model) has undergone both scheduled and interim model updates. These model updates were implemented in June 2001, February 2002, December 2002, May 2003, June 2003, and July 2005. In addition, EGC recently completed a model update during preparation of this response (i.e., October 2007). EGC transmitted the results of the Braidwood Station and Byron Station PRA peer reviews, including disposition of "A" and "B" facts and observations (F&Os), as part of various risk-informed submittals. The transmittal letters for previous submittals of peer review findings to the NRC are listed below.

- Letter from R. M. Krich (Commonwealth Edison Company) to USNRC, "Response to a Request for Additional Information related to 'Request for Amendment to Technical Specifications – Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators'," dated July 7, 2000
- Letter from K. A. Ainger (EGC) to USNRC, "Response to a Request for Additional Information Regarding a Technical Specification Change Request - Extension of Completion Time for Instrument Bus Inverters," dated June 20, 2003
- Letter from Letter from K. A. Ainger to USNRC, "Request for Additional Information Regarding a License Amendment for a One-Time Extension of the Essential Service Water Train Completion Time," dated December 5, 2003
- Letter from D. M Benyak (EGC) to USNRC, "Additional Information Supporting Risk-Informed Inspection Relief Request," dated April 3, 2007

The few remaining open A and B F&Os from the Braidwood Station and Byron Station peer reviews are listed in Table 1. The significance of the open F&Os with respect to the proposed changes is provided in the far-right column. F&Os that have been fully addressed from a technical perspective in the recently completed update no longer have any potential impact on the proposed changes, and are noted as such. These F&Os are still listed in the table since the associated PRA updating requirements evaluation (URE) tracking activities have not yet been closed, as of the date of this response.

<u>NRC Request 3.b</u>
The plant-specific PRA reflects the as-built, as-operated plant.

<u>Response</u>
The Braidwood Station and Byron Station PRA model that was used to validate the applicability of the WCAP-14333 and WCAP-15376 analyses to the RTS and ESFAS instrumentation for each respective station was approved in September 2005, and reflected the as-built and as-operated plants, as of that time. Subsequent to submittal of this license amendment request, EGC approved a revised Braidwood Station and Byron Station PRA model. This new PRA model incorporates recent design and operational changes. EGC has reconfirmed the applicability of WCAP-14333 and WCAP-15376 to Braidwood Station and Byron Station, using data from the recently completed PRA model revision. The impact of plant design or operational modifications that are not reflected in the PRA model that was used to validate the applicability of the WCAP-14333 and WCAP-15376 analyses are discussed in the response NRC Request 3.f below.

From a programmatic perspective, the EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all active EGC nuclear generation sites. EGC transmitted Revision 7 of this procedure to the NRC in a letter dated April 3, 2007, as part of a risk-informed Inservice Inspection Relief Request for Byron Station. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files.

To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, prior to approval of a PRA model, the requirements of the EGC PRA quality assurance programs and procedures that are described in the response to NRC Request 3.d below are followed for model review.

NRC Request 3.c
Applicable PRA updates conducted since completion of individual plant examination (IPE) and IPEEE, and the status of any improvements identified by the IPE and IPEEE.

Response
The response to NRC Request 3.a above delineates the PRA updates that EGC has completed since completion of the initial IPE and IPEEE.

The response to NRC Request 2 above describes the results of the Braidwood Station and Byron Station IPEEE, including the status of any improvements identified by the IPEEE.

The Braidwood Station and Byron Station initial IPE identified a significant risk contribution from internal flooding scenarios at each Station involving a dual unit loss of essential service water from flooding. The risk from internal flooding was substantially addressed and mitigated through several design changes and implementation of internal flooding response procedures, as discussed in the response to NRC Request 2 above. The design changes included the installation of equipment that enables both Stations to use the fire protection system as an alternate means of providing cooling to the centrifugal charging pumps, which, in turn, would prevent loss of all reactor coolant pump (RCP) seal cooling (and the potential for an induced RCP seal loss of coolant accident) following a loss of essential service water. In addition, the discharge of the essential service water system at Braidwood was extended above water level to prevent the potential for draining the cooling lake into the Auxiliary Building, following a rupture of the essential service water system piping. This design change was not needed at Byron as the essential service water system was originally designed with a discharge above water level.

NRC Request 3.d
Reference PRA quality assurance programs/procedures, including expected PRA revision schedules.

Response
The Braidwood Station and Byron Station PRA model is developed, controlled, and maintained in accordance with EGC risk management procedure ER-AA-600. Subordinate EGC procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events (FPIE) PRA models for all active EGC nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on SSCs within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. EGC is currently implementing a new revision to the Braidwood Station and Byron Station PRA model. Therefore, the next planned model update is 2011. The criteria for initiating interim model updates (i.e., updates between the regularly-scheduled updates) are provided in the EGC Risk Management procedures.

NRC Request 3.e

PRA adequacy and completeness with respect to evaluating the proposed CT, bypass test time under Tier 3 configuration risk management.

Response

EGC has reviewed the Braidwood Station and Byron Station PRA model to confirm its adequacy and completeness with respect to Tier 3 evaluations. Significant portions of the reactor trip and engineered safety features actuation signals are modeled in the PRA model. The PRA model is used directly as one input to the Braidwood Station and Byron Station Configuration Risk Management Programs (CRMPs).

The Braidwood Station and Byron Station CRMP is a subset of the work management process, and ensures that configuration risk (i.e., probabilistic and/or deterministic) is assessed and managed, prior to initiating any maintenance activity, consistent with the requirements of 10 CFR 50.65. The CRMP also ensures that risk is reassessed if an emergent condition results in a plant configuration that has not been previously assessed. The CRMP is described in Appendix T of the Braidwood Station and Byron Station Technical Requirements Manual (TRM).

Probabilistic risk assessments of online configurations are performed using the latest approved revision of the level 1 PRA model. Deterministic defense-in-depth evaluations of key safety functions are performed for online and shutdown configurations using a safety function assessment module. Deterministic evaluations of plant configurations that result in a change to initiating event frequency and/or a decrease in mitigation capability are performed using a plant transient assessment module.

The procedures that implement the CRMP at Braidwood Station and Byron Station ensure that representative signals are used, to the appropriate depth, to perform Tier 3 evaluations of the RTS and ESFAS configuration impact. These procedures require that signals that are not explicitly modeled will either be added to the model, or addressed by surrogates that are modeled. This action will result in appropriate modeling in the PRA to complete Tier 3 evaluations. It is expected that in most cases, with one RTS or ESFAS channel unavailable,

risk will be Green, with the exception of those variables modeled in the Reactivity Control safety function deterministic assessment.

NRC Request 3.f

Plant design or operational modifications not reflected in the WCAP-14333/WCAP15376 PRA used in this application that are related to or could impact this application. Justify the acceptability of not including these modifications in the PRA as part of this application.

Response

The Braidwood Station and Byron Station PRA model that was used to validate the applicability of the WCAP-14333 and WCAP-15376 analyses to the RTS and ESFAS instrumentation for each respective station was approved in September 2005, and reflected the as-built and as-operated plants, as of that time.

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. The PRA quality assurance program that is described in the response to NRC Request 3.d above, is utilized for verification and validation of the PRA model.

Subsequent to submittal of this license amendment request, EGC approved a revised Braidwood Station and Byron Station PRA model. This new PRA model incorporates recent design and operational changes. The most significant of these changes was the partial completion of a modification at Braidwood Station that replaced several 120V AC manual transfer switches with automatic transfer switches. This partial implementation of a modification at Braidwood Station improved the reliability of the RTS and ESFAS instrumentation systems. The completion of the modification at Braidwood Station, and the implementation of the modification at Byron Station will also enhance and improve this reliability. In addition, for Byron Station only, the service air compressors were upgraded with air-cooled compressors. This eliminated a support system dependency. The incorporation of this design change had no impact on the reliability of the RTS and ESFAS instrumentation systems.

EGC has reconfirmed the applicability of WCAP-14333 and WCAP-15376 to Braidwood Station and Byron Station, using data from the recently completed PRA model revision. This reconfirmation indicated that the conformance of Braidwood Station and Byron Station with the WCAP acceptance criteria, using the PRA model at that time, bounds the results using the most recent model. The most recent model reflects the current (i.e., October 2007) design and operational configuration for each Station.

NRC Request 4

Consistent with RG 1.174, the cumulative risk of the present TS change in light of past applications (or additional applications planned or under review) should be understood. Cumulative risks were addressed for Byron/Braidwood Stations Units 1 and 2 with respect to the implementation of WCAP-10271, WCAP-14333, and WCAP-15376 only. Provide an evaluation

of cumulative risk impact of previous and additional TS applications under development or review per RG 1.174, Section 3.3.2, as applicable to the implementation of WCAP-14333 and WCAP-15376.

Response
The Braidwood Station and Byron Station risk management processes require that the risk impact of each permanent TS application be evaluated after each periodic PRA update. However, the cumulative risk impact from previous TS applications is not a straightforward calculation, since the risk impacts of the various applications do not necessarily occur simultaneously. In addition, the analyses supporting the various applications were performed with PRA updates reflecting different sets of plant modifications and operating experiences.

Table 2 provides the calculated delta CDF and delta LERF values for each risk-informed TS application from the last complete periodic update. For the reasons noted above, summation of these values is not representative of cumulative risk. However, Table 2 provides the sum of the impacts over all the changes, for comparative purposes.

NRC Request 5

Does the configuration risk management program at Byron/Braidwood provide modeling of the reactor trip and ESFAS systems and components addressed by WCAP-15376 and WCAP-14333 when performing Tier 3 evaluations? Discuss how signals or components not specifically modeled will be addressed.

Response
The response to NRC Request 3.e above describes the Braidwood Station and Byron Station CRMP, with respect to modeling the appropriate reactor trip and ESFAS systems and components for Tier 3 evaluations.

NRC Request 6

Discuss the omission of RCS pressure relief systems from the regulatory commitments listed in Attachment 5, Page 1 of the submittal, see WCAP-15376 (Tier 2).

Response
Upon further review, EGC will revise the two applicable commitments from the original license amendment request (i.e., in a letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, dated January 7, 2007), to include the RCS pressure relief system. The revision of these two commitments is described in Attachment 2. **[Regulatory Commitments]**

Table 1
"A" and "B" F&Os

Cert. Element	Element Description	F&O Level	F&O Description	Status	Resolution	Impact
DA-10	Common cause groups to which the common cause failure probability applies have been derived based on sound judgment and are documented.	B	[Braidwood F&O DA-5] Reviewers do not agree with the justification for asymmetric modeling of the emergency diesel generators (EDGs). [Byron F&O DA-05B] Reviewers do not agree with the rationale provided as a resolution to the Braidwood finding.	Resolved in Rev. 6 of the Model	The asymmetric modeling assumption was removed and the model logic was changed to consider symmetric common cause among all four EDGs (two for each plant).	None
HR-2	Human reliability analysis (HRA) is consistent with industry practice.	B	[Braidwood F&O HR-2] The modified cause based decision tree (CBDT) method used for the HRA apparently deviates from standard industry practice.	Resolved in Rev. 6 of the Model	Subsequent updates to the HRA methodology and documenation eliminated the conflicts between CBDT and human cognitive reliability modeling. The CBDT methods from Quad Cities were used, which subsequently received favorable peer review certification.	None

Table 1 "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**Impact**
HR-10, 22	Assessment of plant procedures and plant specific operating experience are explicitly included in the identification and quantification process for the human interactions (HIs). The models and analysis are consistent with the operating procedures and training.	B	[Braidwood F&O HR-4] The steam generator tube rupture (SGTR) event tree structure and HRA do not reflect the circumstances around entering FR-H.1. The success criteria, event tree structure, and HRA should be modified to reflect accident sequence mitigation dictated by the emergency operating procedures (EOPs).	Resolved in Rev. 6 of the Model	Operator interviews were conducted as part of the Rev. 6 changes to the HEPs. Entry inot FR-H.1 was confirmed and timing of actions was verified.	None

Table 1
"A" and "B" F&Os

Cert. Element	Element Description	F&O Level	F&O Description	Status	Resolution	Impact
HR-11	The symptoms available during the postulated accident sequence are evaluated and input into the HRA process.	B	[Byron F&O HR-07B] The HRA for manual Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) actions do not explicitly account for degradation of plant monitoring (i.e., different HEP depending on whether or not failure of auto-actuation of equipment was due to equipment failure or signal failure).	Resolved in Rev. 6 of the Model	The manual reactor trip operator action is conditioned on the basis of whether the failure was due to actuation logic failure versus signal failure. However, the operator action to initiate ESFAS manually was not modified. The Braidwood and Byron Station procedures require manual actuation of Safety Injection (SI) regardless of whether or not signals are present. In addition, the procedures require manual actuation of individual components if auto actuation fails, regardless of the status of signal inputs. The HEPs in the model reflect both the SI actuation and the individual component actuation as well as the conditional probability between the two.	None

Table 1
"A" and "B" F&Os

Cert. Element	Element Description	F&O Level	F&O Description	Status	Resolution	Impact
HR-11, 14, 18, 20	The symptoms available during the postulated accident sequence are evaluated and input into the HRA process. Operator actions have been reviewed by the operating staff and their impact is included in the HRA evaluation The performance shaping factor for time available for an action and the time required to take an action are developed on a plant specific basis. The time required to complete the actions is based on observation or operations staff input.	B	[Byron F&O HR-02B] Although a significant effort has been undertaken to gain operator input into evaluating HEPs, the reviewers found the effort to date had not fully addressed observations from the Braidwood peer certification (F&O HR-5). Specifically cited is the lack of operator interviews to validate input assumptions, timing and logic of certain key operator actions. The reviewers also acknowledged that the sensitivity analyses performed showed that the overall results are not overly sensitive to the operator action modeling.	Resolved in Rev. 6 of the Model	As part of the Rev. 6 model revisions, all significant operator actions were reviewed with Operations and Training personnel to assess the timong for execution and cognition, as well as the procedural guidance for the actions.	None

Table 1
"A" and "B" F&Os

Cert. Element	Element Description	F&O Level	F&O Description	Status	Resolution	Impact
QU-27, 28	A search is performed for unique or unusual sources of uncertainty not present in the typical or generic plant analysis. If there are unusual sources of uncertainty, special sensitivity evaluations or quantitative uncertainty assessments are performed to support the base conclusion and future applications.	A	[Braidwood F&O QU-7] Only parametric uncertainty analyses have been performed.	Open	Formal guidance for addressing this issue is still being developed by NRC and EPRI.	The circumstance described by this WOG PEER review finding is not expected to adversely affect the conclusions made for the proposed RPS and ESFAS extended surveillance test interval (STI) and completion times (CTs) because of the conservatism used in the Braidwood Station and Byron Station PRA model. In addition, this PRA model is not used to calculate changes in CDF, LERF, or ICCDP to justify the proposed changes.

Table 2 Estimated Cumulative Risk				
Item no.	**LAR Application Date**	**Calculated ΔCDF (internal events)**	**Calculated Δ LERF (internal events)**	**LAR**
1	9/1/2000	Small decreases in risk for both Bryon and Braidwood with the compensatory measure credited.		Emergency Diesel Generator CT Extension
2	11/19/2003	7E-10 Byron 9E-10 Braidwood	4E-09 Byron 7E-09 Braidwood	Inverter CT Extension
3	2/14/2006	9E-08 Byron 1E-07 Braidwood	1E-09 Byron 1E-09 Braidwood	Risk Informed ISI Relief Request
4	4/7/2007	N/A	2E-07 Byron 2E-07 Braidwood	ILRT STI extension (i.e., from once in 10 years to once in 15 years)
5	1/7/2007	< 1E-06 Byron < 1E-06 Braidwood	< 1E-07 Byron < 1E-07 Braidwood	RTS/ESFAS Instrumentation CT and STI Extension
6	Planned Submittal: Early 2008	< 1E-08 Byron < 1E-08 Braidwood	< 1E-08 Byron < 1E-08 Braidwood	RCP Flywheel
Approximate Sum Cumulative Risk		**1E-06 Byron 1E-06 Braidwood**	**3E-07 Byron 3E-07 Braidwood**	

The following table identifies those actions committed to as part of the EGC response an NRC Request for Additional Information that was provided in a letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), dated August 17, 2007.

The first commitment applies to Byron Station and is a new commitment. The two remaining commitments apply to both Braidwood Station and Byron Station and are revisions of the same commitments that were made in the original license amendment request. Any other statements in this submittal are provided for information purposes and are not regulatory commitments.

Commitment	**Committed Date**	**Due Date/Event**	
		One-Time Action (Yes/No)	**Programmatic (Yes/No)**
EGC will implement enhanced internal flood response procedures at Byron Station.	Prior to implementation of the revised TS.	No	Yes
Revision of Original Regulatory Commitment EGC will implement administrative controls to ensure that activities that degrade the availability of the RCS pressure relief system, auxiliary feedwater system (AFW), AMSAC, or turbine trip should not be scheduled when a logic train or an RTB train is inoperable for maintenance.	Prior to implementation of the revised TS.	No	Yes
Revision of Original Regulatory Commitment EGC will implement administrative controls to ensure that activities that result in the inoperability of electrical systems (e.g., AC and DC power) and cooling systems (e.g., essential service water and component cooling water) that support the RCS pressure relief system, AFW system, AMSAC, turbine trip, one complete train of ECCS, and the available reactor trip and ESFAS actuation functions should not be scheduled when a logic train or an RTB train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available	Prior to implementation of the revised TS.	No	Yes