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Nuclear

RS-07-050

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

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

Subject: Additional Information Supporting Risk-Informed Inservice Inspection Relief Request

References:
1. Letter from D. M. Hoots (Exelon Generation Company, LLC) to U. S. NRC, "Byron Station, Units 1 and 2, Transmittal of Inservice Inspection Program Plan for the Third Ten year Inspection Interval," dated February 14, 2006
2. Memorandum from C. Gratton (U. S. NRC) to M. L. Marshall (U. S. NRC), "Byron Station, Unit Nos. 1 and 2 – Facsimile Transmission of Draft Request for Additional Information (TAC Nos. MD3855 and MD3856)," dated February 26, 2007

In Reference 1, Exelon Generation Company, LLC (EGC) submitted the third ten-year inspection interval Inservice Inspection Program for Byron Station, Units 1 and 2. Section 8 of the Inservice Inspection Program plan contained alternatives to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inspection and Testing of Components of Light Water Cooled Plants." Relief Request I3R-02 requested NRC approval to implement alternative risk-informed selection and examination criteria for certain pressure retaining piping welds.

On January 31, 2007, the NRC provided a draft request for additional information (RAI) related to risk-informed Inservice Inspection Program Relief Request I3R-02. The draft RAI was clarified in a conference call between EGC and the NRC on February 8, 2007. The results of the February 8, 2007, conference call are documented in Reference 2. In response to this request, EGC is providing the attached information.

There are no regulatory commitments contained in this letter. Should you have any questions related to this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Darin M Benyak

Darin M. Benyak
Manager, Licensing

Attachments:
1. Response to Request for Additional Information
2. ER-AA-600-1015, "FPIE PRA Model Update," Revision 7

cc: NRC Senior Resident Inspector
NRC Regional Administrator, Region III

ATTACHMENT 1
Response to Request for Additional Information

NRC Request 1

The last paragraph on page 3 of 5 of relief request I3R-02 states that, "[t]he Consequence Evaluation, Degradation Mechanism, Risk Ranking, and Element Selection steps encompass the complete living program process applied under the Byron RI-SI program." Are the inspection locations in the RI-ISI program that has been developed for the third interval the same locations as those in the program approved in the NRC staff's February 5, 2002, safety evaluation? If not, please summarize the changes to the program and what caused those changes.

Response

As a "living program," the Risk-informed Inservice Inspection (RISI) Program methodology requires on-going revisions due to changes that occur after the original implementation. The following four items describe situations where the initially selected welds were replaced, added, or deleted due to changes in the RISI Program, or where unacceptable scanning limitations were determined at the time of the weld examination.

Item 1: Changes in selection due to limited access to the examination surface

Weld configurations, such as pipe-to-valve or adjacent obstructions, may present limited coverage under the examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inspection and Testing of Components of Light Water Cooled Plants," Appendix VIII. Reselection of some initial examination locations was required where greater than 90% coverage could not be obtained.

Item 2: Changes in selection due to reclassification into different RISI categories

Revision of the RISI Program with the updated probabilistic risk assessment (PRA) model resulted in some segments being reclassified into different RISI categories. Examples include the following.

Main Steam system piping was initially assigned to Category 6 (i.e., low failure potential/medium consequence) requiring no examination selections, and was later revised to Category 4 (i.e., low failure potential/high consequence) requiring 10% examination selection.

Some Safety Injection piping segments were initially assigned to Category 6 (i.e., medium failure potential/low consequence) requiring no examination selections, and were revised to Category 5 (i.e., medium failure potential/medium consequence) requiring 10% examination selection.

Other Safety Injection piping segments were initially assigned to Category 4 or Category 5 requiring 10% examination selection. The Category 5 segments were changed to low consequence, which resulted in a reclassification to Category 6 requiring no examinations. The Category 4 segments were changed to medium consequence, thus becoming Category 6 segments requiring no examinations. In both cases, the welds selected for examination were adjusted.

Item 3: Changes due to reassessment of degradation mechanism

Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," was used as the basis of Byron Station's RISI Program. The evaluation of primary water stress corrosion cracking (PWSCC) of Alloy 600 materials specifies a minimum temperature of 620°F for inclusion of this degradation mechanism. Based on input from the Materials Reliability Program (MRP-139), Byron Station altered the PWSCC evaluation to include all Alloy 600 components. This resulted in weld reclassification from Category 4 (i.e., low failure potential/high consequence) requiring a 10% examination selection to Category 2 (i.e., medium failure potential/high consequence) requiring a 25% examination selection.

Item 4: Lines added with new ASME Code requirements

For ASME Class 2 components, the IWC-1220 exemption criteria were revised to require the examination of smaller size piping in the Auxiliary Feedwater (AF) system. AF piping was evaluated for degradation and consequence along with other non-exempt piping systems. The AF piping segments were classified into Category 2 or Category 4 as appropriate. AF welds were selected per the category requirements of 25% or 10% respectively. EGC's response to NRC Request 5 below provides additional information regarding the selection of AF welds.

The following tables provide a summary of the changes to RISI inspection population for Byron Units 1 and 2.

BYRON UNIT 1			
RISK CATEGORY	**EXAMS (RISI REV. 0)**	**EXAMS (RISI REV. 5)**	**ITEMS AFFECTING CHANGES (SEE ITEMS 1 – 4 ABOVE)**
High	85	111	▪ Limited Exam Coverage ▪ Degradation Mechanism Update ▪ New Scope due to ASME Code
Medium	168	221	▪ Limited Exam Coverage ▪ RISI Category Reclassification ▪ Degradation Mechanism Update ▪ New Scope due to ASME Code
Total	**253**	**332**[1]	
[1] 14 additional welds have been added to the RISI inspection population due to the inclusion of the Break Exclusion Region (BER) piping in Revision 4 to the RISI Program. The total inspections currently scheduled under Revision 5 of the RISI Program is thus 346.			

BYRON UNIT 2			
RISK CATEGORY	**EXAMS (RISI REV. 0)**	**EXAMS (RISI REV. 5)**	**ITEMS AFFECTING CHANGES (SEE ITEMS 1 – 4 ABOVE)**
High	112	127	▪ Limited Exam Coverage ▪ Degradation Mechanism Update ▪ New Scope due to ASME Code
Medium	164	191	▪ Limited Exam Coverage ▪ RISI Category Reclassification ▪ Degradation Mechanism Update ▪ New Scope due to ASME Code
Total	**276**	**318**[2]	
[2] 17 additional welds have been added to the RISI inspection population due to the inclusion of the BER piping in Revision 4 to the RISI Program. The total inspections currently scheduled under Revision 5 of the RISI Program is thus 335.			

NRC Request 2

Paragraph 4 on page 3 of 5 of your I3R-02 submittal states, "the original risk impact assessment is not a necessary element of the implementing process and is not required to be continually updated." The change in risk acceptance guidelines must continue to be met as the facility, PRA, and the risk-informed program change over time. Please provide the risk impact of implementing the RI-ISI program proposed for the third interval instead of the American Society of Mechanical Engineer's inspection program that was replaced by risk-informed inservice inspection. (The NRC staff has concluded that it is an unnecessary burden to develop a new ASME inspection program when transitioning to a newer edition of the ASME code for the sole purpose of estimating the risk impact of a RI-SI program)

Response

As part of updating the RISI analysis for the third 10-year interval, the original risk impact assessment was also updated to confirm the change in risk was maintained within the acceptance guidelines. The original methodology of the calculation was not changed, and the change in risk was simply re-assessed using the initial 1989 Section XI program prior to RISI and the new element selection for the third 10-year interval RISI program. This same process has been maintained in each revision to the Byron RISI Report that has been performed to date.

Using this process, the change in risk for Unit 1 was 9.21E-8 for delta-core damage frequency (delta-CDF) and 1.45E-9 for delta-large early release frequency (delta-LERF). For Unit 2, the values were 5.78E-8 for delta-CDF and -3.53E-10 for delta-LERF. These values are all within the 1.00E-6 and 1.00E-7 acceptance criteria for delta-CDF and delta-LERF respectively. The change in risk analysis was likewise done at a system level, and no system acceptance criteria are exceeded in the current program using the latest RISI element selections.

NRC Request 3

According to your submittal dated November 17, 2000, and the NRC staff's safety evaluation dated February 5, 2002, the Byron probabilistic risk assessment (PRA) had not undergone a peer review prior to development of the second interval's RI-ISI program. Instead, the submittal states that you used the results of the peer review on the sister plant Braidwood Station, Units 1 and 2, to address potential PRA quality issues. Please indicate when the peer review was performed on the Byron PRA. Please provide the A and B level facts and observations from the Byron PRA peer review and the resolution of each observation or an explanation about why resolving the observation is not expected to significantly affect the proposed RI-ISI program.

Response

The Braidwood PRA was subjected to a Westinghouse Owners' Group peer review in September 1999. The Byron PRA was subjected to a separate 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 and Byron PRA models are very similar and exist in an integrated model) has undergone major upgrades as well as interim upgrades (i.e., in January 2002, August 2004, and March 2006).

A list of the open A and B facts and observations (F&Os) remaining from the Byron and Braidwood peer reviews is provided below. Because the models are integrated, the outstanding significant F&Os are listed from each site. The importance of the finding with regards to the RISI application is listed in the right-most column.

The importance of the findings is evaluated in the context of how the PRA is used for the RISI application. A weld inspection regime is based on two variables: (1) PRA risk significance, and (2) susceptibility to damage mechanisms. The following table illustrates the relationship and the associated inspection importance category.

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	**CONSEQUENCES OF PIPE RUPTURE** IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBIBILITY			
	NONE	**LOW**	**MEDIUM**	**HIGH**
HIGH FLOW ACCELERATED CORROSION	**LOW** Category 7	**MEDIUM** Category 5	**HIGH** Category 3	**HIGH** Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 5	**HIGH** Category 2
LOW NO DEGRADATION MECHANISMS	**LOW** Category 7	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 4

The pipe rupture consequences for Byron were evaluated using the PRA model and are in terms of conditional core damage probability (CCDP) and conditional large early release probability (CLERP). According to the approved methodology, the numerical results of the analysis are then binned into one of three PRA consequence categories: High, Medium, and Low.

CCDP AND CLERP VERSUS CONSEQUENCE CATEGORIES		
CONSEQUENCE CATEGORY	**CCDP RANGE**	**CLERP RANGE**
HIGH	CCDP > 1E-4	CLERP > 1E-5
MEDIUM	1E-6 < CCDP ≤ 1E-4	1E-7 < CLERP ≤ 1E-5
LOW	CCDP ≤ 1E-6	CLERP ≤ 1E-7

Given this binning process, the results of the PRA, in most cases, would have to change by at least an order of magnitude in order for a weld to experience a change in inspection regime. Consequence results near the thresholds do not have to change as much. On the other hand, consequences <<1.0E-6 CCDP (and <<1.0E-07 CLERP) can experience much larger changes with still no impact on the weld inspection program.

As a result of these peer reviews and responses to the open A and B F&Os, the PRA is considered to meet the PRA quality criteria of Regulatory Guide 1.174.

Open "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**RISI 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.	Mostly Resolved	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). However, not all the EDG basic event calculations were done correctly. Hence, the F&O is not closed.	Negligible. Many of the RISI consequences prescribe an initiator as a result of the assumed pipe break. Multiple EDGs are only used for loss of offsite initiators which are not likely coincidental with a pipe break. Induced loss of offsite power (LOOP) was not part of the PRA model used for RISI, but has been adopted in interim revisions. The induced LOOP cutsets generally contain the basic event for failure of all four EDGs. This would have yielded conservative results and if corrected would likely reduce the CCDP/CLERP of some consequences.
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.	Mostly Resolved	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. However, the flooding analysis human error probabilities (HEPs) used a modified CBDT that has some differences between the other HEPs and should be reviewed.	Negligible. The remaining HEPs with differences from the industry model response to internal flooding initiating events. These HEPs neither model mitigation of pipe breaks stemming from weld failures, nor operator response to initiating events caused by weld failures. The cutsets with these HEPs form part of the base model and are subtracted out in the CCDP/CLERP calculation for degradation of a mitigating system.

Open "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**RISI 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).	Partially Resolved	Model assumes that procedures will direct operators to enter FR-H.1 and use bleed and feed during a SGTR with a loss of AF. However, it is not clear from the procedures that this is what the operators would do. The same timing is used for SGTR as is used as for a complete loss of feedwater. However, this is an assumption and is not based on walk-through or talk-through of the SGTR procedures.	Negligible. An evaluation of the operator action and timing was performed as part of the mitigating systems performance index (MSPI) documentation. The analysis indicated that it is likely the operators will proceed in a manner consistent with the HRA. However, SGTR, like internal flooding, is not an initiating event expected to occur as a result of a weld failure. The cutsets with this HEP form part of the base model and are subtracted out in the CCDP/CLERP calculation for degradation of a mitigating system.
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).	Partially resolved	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. Separate operator actions are modeled to manually initiate the Safety Injection (SI) signal, as well as to manually actuate components, given successful SI initiation with failure of the component to actuate (e.g., such as due to failure of an actuation relay). Thus, only the manual SI actuation HRA should be affected by the potential failure of indication (e.g., sensor signals).	Minimal. A sensitivity was performed where the HEP to manually initiate SI was set to true in the saved cutsets. Only the CCDP of large and medium loss-of-coolant accidents (LOCAs) changed appreciably. These PRA consequences are already high. None of the PRA consequence evaluations changed bins (e.g., low to medium or medium to high PRA consequence).

Open "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**RISI 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.	Partially Resolved	Operator interviews were conducted on significant operator actions following the certifications, including validation of the timing for actions specified in the F&O. A review of the risk significant operator actions was performed. All but three of those actions had operator input.	Negligible. Three risk significant operator actions did not appear to have documented operator input into the timing assumptions. One action is concerned with responding to a SGTR, which is not relevant to RISI. One action is concerned with mitigating a stuck open power operated relief valve (PORV) with the block valve. Upon further analysis, if the available time for the action is reduced to five minutes, the existing probability remains conservative. The last action addresses termination of injection prior to challenging the PORV. A sensitivity was performed assuming a screening value of 0.1 on top of the sensitivity of HR-11 above. The largest additional change was only 2%. No PRA consequence would have changed rank.

Open "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**RISI 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	A set of sensitivity analyses was performed to examine a number of issues associated with the use of the PRA to support the EDG Technical Specification changes and to address specific issues raised by the certification team, including: Thermal hydraulic analyses to investigate different reactor coolant pump (RCP) seal leak initiation times for "popping failure modes" and their impacts on the electric power recovery split fractions. Thermal hydraulic analyses to investigate the impact of steam generator water volume assumptions on steam generator dry out times and impacts on the HRA values for bleed and feed actions. Impact of compensating measures on all the risk metrics used to evaluate the EDG Technical Specification changes. Impact of alternate initial plant configurations on these same risk metrics. Impact of different plant strategies to utilize the increased EDG completion time for unplanned maintenance. Impact of not crediting a variety of operator recovery actions. Throughout the HRA task, numerous sensitivity studies were performed to evaluate the impact on derived HEPs.	None. Based on the activities completed to date, the fundamental issues have been addressed; however, this F&O is considered open until a roadmap is documented to identify the process used and results of the search for unusual uncertainties. The absence of the supporting documentation has no impact on the current RISI analysis.

Open "A" and "B" F&Os						
Cert. Element	**Element Description**	**F&O Level**	**F&O Description**	**Status**	**Resolution**	**RISI Impact**
TH-9	Documentation provides the basis of the thermal hydraulic analysis, is traceable to plant specific or generic analysis, and demonstrates the reasonableness of the success criteria.	B	[Braidwood F&O TH-3] It is difficult to match success criteria to specific analyses and fault trees.	Resolved	Success criteria are described in the Success Criteria Notebook with the specific supporting analyses noted. The Event Tree Analysis Notebook also adequately references success criteria. The peer review finding also calls for cross referencing success criteria to fault tree gates or event tree headings.	None. This is a documentation issue only, and the suggested resolution exceeds the ASME standard requirements. However, a table of Success Criteria for equipment functions was created to address this issue and currently resides in the MSPI program document. The RISI analysis and PRA are up-to-date with regard to system function success criteria and its associated documentation. Therefore, this item is closed.

NRC Request 4

Risk-informed applications should be developed using a technically adequate PRA that is based on the as-built, as-operated, and as-maintained plant. Please provide the following:

(a) The revision name or number, date, and base core damage frequency (CDF) and large early release fraction (LERF) of the Byron PRA model used to perform the risk ranking of pipe segments and change in risk evaluation in preparation for the third 10-year ISI interval.

(b) Please provide a summary of how the changes to the PRA are developed and reviewed.

Response

The Byron PRA model revision used to perform the risk ranking of pipe segments and to evaluate the consequences of pipe rupture for the RISI assessment is documented in BB PRA-014, "Quantification Notebook, Byron and Braidwood Stations," Revision 5B, Addendum 1, dated April 2004. The base CDF and base LERF from the Byron PRA model for Unit 1 are 6.1E-05 per year and 4.7E-06 per year, respectively. For Unit 2, the base CDF is 6.1E-05 per year and base LERF is 5.5E-06 per year.

An EGC PRA maintenance and update procedure, ER-AA-600-1015, "FPIE PRA Model Update," formalizes the PRA update process. The procedure defines the process for regular and interim updates for tracking issues identified as potentially affecting the PRA, and for controlling the model and associated computer files. Attachment 2 provides a copy of ER-AA-600-1015.

NRC Request 5

The newer versions of the ASME Code have reduced the exempted portions of Auxiliary Feedwater piping from nominal pipe size (NPS) 4 to NPS 1½. This reduction in exempted piping has caused other licensees to add ASME Class 2 and/or Class 3 Auxiliary Feedwater piping to the scope of their RI-ISI programs, and to implement their chosen RI-ISI methodology to classify, risk-rank, and to select, as necessary, additional locations for the next ISI interval. Please describe how you treated this issue in your RI-ISI program for the third 10-year ISI interval when you updated your code of record from the 1989 edition to the 2001 edition with 2003 addenda.

Response

The RISI Program is applied to ASME Class 1 and 2 piping systems for both Byron Units 1 and 2. Within those systems, the RISI Program applies to the portion of piping not exempted by IWB-1220 and IWC-1220 respectively. AF piping was evaluated for failure potential and consequence of failure along with other non-exempt piping systems. The AF piping segments were classified into the appropriate RISI categories and elements were selected per the category requirements for examination during the third inspection interval. The inclusion of the new AF welds into the RISI analysis resulted in the selection of five additional Category 2 AF welds and nine additional Category 4 AF welds in each of the Unit 1 and Unit 2 examination populations.

ATTACHMENT 2

ER-AA-600-1015,
"FPIE PRA Model Update,"
Revision 7

FPIE PRA MODEL UPDATE

1. PURPOSE

1.1. This T&RM establishes responsibilities and general guidelines for updating the full power, internal events (FPIE) Probabilistic Risk Analysis (PRA) Models at all active nuclear generation sites.

2. TERMS AND DEFINITIONS

2.1. **Guidance** – Guidance in the context of this T&RM is a means of accomplishing procedural and regulatory requirements. It does not preclude accomplishment of these requirements by other means

2.2. **MAAP (EPRI Modular Accident Analysis Program):** A thermal-hydraulic computer code utilized to determine plant-specific response under postulated severe accident scenarios. Provides information such as time for core coolant boil off, time for core melt and RPV breach, pressures and temperatures in the RPV and reactor building areas, and the time to reach these pressures and temperatures.

2.3. **PRA Maintenance:** PRA maintenance involves the collection and evaluation of new information which could impact the PRA model and updating the model and applications as appropriate.

2.4. **PRA Periodic Update:** Revision of the PRA and associated documentation involving evaluation of the adequacy of all technical elements of the PRA on a specified schedule. This includes ASME RA-S-2002 (Ref. 6.2) PRA maintenance and upgrade attributes.

2.5. **PRA Unscheduled Update:** Revision of the PRA to incorporate a change of plant design or operation having sufficient impact to the results that it should not wait to the next periodic update or a PRA revision required to correct a PRA model error which should not wait to the next periodic update.

2.6. **PRA Update Project Plan:** The Project Plan is a document describing the PRA update tasks, personnel and resource requirements, and schedule.

2.7. **Proper Software:** Software meeting the requirements of IT-AA-101 (ref.6.3) or its equivalent.

2.8. **Rollout:** The review and revision of PRA applications and documentation, publicizing of results after a PRA model update.

2.9. **Unavailability:** The unavailability of a component or system is the fraction of time that a system or component is not capable of supporting its function including but not limited to the time it is disabled for test or maintenance.

2.10. **URE (Updating Requirement Evaluation):** An evaluation in which it is decided whether the change item being evaluated (hardware item or administrative item) requires a revision to the current plant PRA model. The evaluation includes scheduling required model revisions based on the significance of the PRA impact. See Attachment 1 for an example of a URE form. The URE form may deviate from Attachment 1, but should contain the key elements contained in the example.

2.11. **Acronyms**

CDF – Core Damage Frequency

FPIE – Full Power, Internal Events

F-V – Fussell-Vesely (importance measure)

LERF – Large Early Release Frequency

MAAP – Modular Accident Analysis Program

NFM – Nuclear Fuels Management

PRA – Probabilistic Risk Analysis

PSA – Probabilistic Safety Analysis

RAW – Risk Achievement Worth

RPV – Reactor Pressure Vessel

3. RESPONSIBILITIES

Risk Management consists of an integrated team where individuals assigned specific roles and responsibilities delineated below will perform the actions of their position regardless of their employer.

NOTE: Contractors (This note is to address use of personnel outside of the out-sourced contractor and Exelon organization) may perform any of the tasks stated in sections 3.2 or 3.3 if approved by the Senior Manager Risk Management

3.1. Senior Manager Risk Management

3.1.1. Maintains a Periodic Update schedule for all Nuclear Stations.

3.1.2. Interfaces with Station personnel to obtain concurrence with the Project Plan.

3.1.3. Incorporates update schedule into non site RM work management plan (reference ER-AA-600-1011).

3.1.4. Reviews qualification of personnel assigned to update tasks and assures experience and/or training is sufficient to support successful completion of those tasks.

3.2. Model Owner

3.2.1. Maintain a URE database for each site's PRA.

3.2.2. Establish a Project Plan for the update which identifies tasks to be performed, a schedule to complete each task and resources to carry out the tasks.

3.2.3. May update the plant PRA model according to the Project Plan.

3.2.4. Maintain list of key site procedures.

3.3. Site Risk Management Engineers

3.3.1. Be actively involved with PRA updates, performing update tasks as part of the update team.

3.3.2. Provide general PRA training and application specific training to site personnel such as Site Engineers.

3.3.3. Creates change management plan for model update rollout activities.

3.3.4. Perform independent reviews of PRA updates performed for other sites as assigned.

3.3.5. May initiate and/or disposition UREs as specified in this T&RM.

4. MAIN BODY

4.1. Update Intervals

4.1.1. **PREPARE** periodic updates for each station PRA model and associated PRA model Category 1 documentation (reference 6.4, ER-AA-600-1012) on a schedule agreed upon by the Site Engineering Director, corporate Design Engineering Director, and Senior Manager Risk Management. The typical interval between PRA updates is 4 years. If the periodic update interval will exceed four years for any PRA technical element, **DOCUMENT** justification that the PRA continues to represent adequately the as-built, as-operated plant.

4.1.2. **CONSIDER** including the following items (at a minimum) in the periodic update:

- Design Changes
- Procedure Changes
- Technical Specification Changes
- Component Failure Rates
- Component Maintenance Unavailability
- Initiating Event Frequencies

- Changes to Design Basis Calculations and or Assumptions
- Open UREs
- Changes to PRA technology
- Industry experience
- Site operating Experience
- Revisions to PRA standards
- ASME standard gap analysis

4.1.3. In addition to periodic updates, the need for unscheduled updates (PRA maintenance) may arise. A Risk Management Engineer will evaluate each URE prepared to determine whether the current PRA model should be immediately updated or the update can be delayed to the next scheduled update. The evaluation will be documented in the URE database. This determination will be made based on whether the PRA model fidelity (representation of the as-built, as operated plant) without the update is adequate to support current PRA applications.

1. UREs will be evaluated to determine the need for an unplanned update within 30 days of creation.

4.1.4. An unscheduled update may also be required if an error is identified in the PRA model.

1. A PRA model error requiring an unscheduled update is one that affects the results of the PRA in a fashion that can or does affect applications. For example a fault tree error that would cause a MOVs classification in the MOV program to change would have to be corrected as would an error that would change the FV of a MSPI monitored component beyond allowable bounds. An error that does not affect the conclusions may be placed in a URE for the next periodic update. An example would be a basic event description that is wrong but the basic event is appropriately handled in the model. Also if there is a large impact on CDF or LERF such as a 25% change or a significant shift in distribution, an unscheduled update should be considered.

4.1.5. If a URE requires an unplanned update of the PRA model, an IR should be generated if one does not already exist. Additionally if an URE has an impact on current applications, an IR should be generated.

4.2. <u>Periodic Update Process</u>

4.2.1. <u>Project Planning</u>

1. The model owner will prepare a Project Plan for the periodic updates which identifies tasks to be performed, a schedule to complete each task and resources to carry out the tasks. The schedule should include the performance of the update, the reviews and approvals and the completion of documentation. The schedule should be agreed to by the Senior Manager Risk Management.

2. Consider the need to update other PRA models such as fire and seismic. If update of non-FPIE PRA models is deferred, document the basis and create UREs specific to those models.

3. The SRME will create change management plan per HU-AA-1101 for the model update roll out activities within 30 days of the approval of the PRA model.

4. The progress of the update and roll out should be periodically statused by the model owner, SRME and Senior Manager Risk Management

4.2.2. Data collection

1. OBTAIN AND REVIEW data and information from the following sources:

   A. Equipment Performance Data (reliability & availability) based on Maintenance Rule Records and CDE (MSPI) records supplemented by Condition Reports and Out of Service Records, as necessary

   B. Operating procedures that were used to determine operator actions

   C. Surveillance test and operating procedures for changes in the test frequency, test duration, and other aspects that are applicable to failure rate calculations, demands and run hours

   D. New and revised Design Changes, Technical Specifications and Design Calculations to determine those changes that require modeling changes in order to represent adequately the as-built, as-operated plant

   E. Unit Availability Data (normally available from CDE) supplemented by Event Reports to determine if an update to the Initiating Event Database is required

   F. Site operating experience from CRs, NERs, etc.

   G. Open UREs to determine those that will be included in the update. Changes that most significantly impact risk informed applications should be included in the next periodic update.

2. In the performance of a periodic update, **OBTAIN** the concurrence of the Senior Manager Risk Management for UREs deferred to the next update.

3. The above data should also be evaluated for impact on other PRA models such as fire and seismic PRA.

4. **DOCUMENT** the results of the review in a manner that allows determination that a specific change has been reviewed and dispositioned. This may be included in the documentation category 2 reports for the update.

5. **REVIEW** industry practices and **DETERMINE** if changes in industry "state-of-the-art" data collection and analysis practices should be applied.

4.2.3. Model revision

1. **UPDATE** the PRA technical elements that require changes in order to represent adequately the as-built, as-operated plant. **CONSIDER** the following minimum set of elements during the update:

   A. Failure Rates-

   At a minimum consider updating failure rates and maintenance unavailability rates for components/trains with RAW > 2.0 or F-V > 0.005 in the current model. **DOCUMENT** justification for any risk-significant components that will not be updated or for which only generic failure rates will continue to be used

   B. Maintenance Unavailabilities

   Update with plant data to data cutoff date

   Evaluate need to model concurrent maintenance unavailabilities.

   C. Fault Trees that are significantly impacted by plant modifications

   D. Event Trees that are significantly impacted by plant modifications and/or revisions to Operating Procedures

   E. Initiating Event Data

   Update with plant data to data cutoff date

   F. Thermal-hydraulic analyses (MAAP) that are significantly impacted by new calculations or revisions to Operating Procedures.

   G. Human Error Probabilities that are significantly impacted by revisions to Operating Procedures or Policies

4.2.4. Quantification and Review

1. **QUANTIFY** the model.

2. **PERFORM** a review of the updated model results including the items in Attachment 2 prior to final approval of the model.

3. Have the appropriate system manager(s) review system related changes especially assumptions.

4. Have operations training or Operations review changes to HEPs especially assumptions

5. Review the updated model against the appropriate sections of Reference 6.2. Appropriate sections are those that were altered by the PRA model update. For example if initiating events were updated or the method for calculating them changed, the initiating events section of the ASME Standard should be reviewed to ensure the updated model still meets the necessary elements.

    A. Some changes may require at least a limited peer review to be performed per the ASME PRA standard.

4.2.5. Approval

The updated PRA model will be considered approved when the quantification notebook (level 1 CDF and level 2 LERF) is approved. At this time the PRA may be used for applications.

The Summary notebook should be approved as soon as possible after the quantification notebook.

The PRA is considered complete when all required applications and supporting documentation such as system notebooks are updated. Completion should occur no later than six months after approval. If completion of the supporting documentation will not be completed within 6 months, approval of the Senior Manager, Risk Management is required.

4.2.6. Documentation

PREPARE a CDF and LERF documentation category 1 quantification notebook at each periodic update. Include in the summary notebook the changes made to the model. Store the model and documentation in accordance with ER-AA-600-1014 (ref. 6.6).

4.3. **Rollout**

4.3.1. The SRME creates a change management plan per HU-AA-1101 within 30 days of the approval of the PRA model for the below actions at a minimum.

4.3.2. **EVALUATE** at least qualitatively all current documentation category 1 documents which are affected by the periodic PRA update and **DETERMINE** whether revision is necessary.

The following should be evaluated for revision needs after every periodic update:

A. PRA Summary Report.

B. Training Aids & Posters.

C. Appropriate management briefings and training to assure plant personnel are kept apprised of new insights gained or revised importance measures.

D. The CDF and LERF baselines for use in trending and other applications

E. The On-Line Risk Monitor

F. The MSPI basis document (See below)

G. PRA Model Category 2 documentation (ref. 6.5, ER-AA-600-1012)

H. List of risk-significant systems/components for input to the Maintenance Rule Expert Panel

I. Component risk rankings as required (for example MOVs, and AOVs)

J. The analysis for acceptability of the Maintenance Rule Performance Criteria (ER-AA-600-1044)

K. (If appropriate) risk informed ISI supporting analyses (schedule may be set by ISI program manager).

L. Equipment importance lists for applications.

M. Notify Security of new PRA base model

N. List of procedures that impact the PRA (procedures which the site RME reviews quarterly) and add any arising from the update activities.

O. All current PRA applications, and **SCHEDULE** revisions as appropriate. A qualitative review is acceptable if it clearly demonstrates that there is no significance impact on a current application.

P. <u>Limerick only</u>

Surveillance Frequency Control Program surveillance test interval evaluations

4.3.3. The MSPI Basis Document must be updated in the quarter following approval of the PRA model. See ER-AA-600-1047 (Ref. 6.7) for additional detail.

4.3.4. **DOCUMENT** the review of PRA applications, including those where there is no impact. Any PRA application whose ER-AA-600-1012 documentation states, "not required to be updated" or whose PRA applications listing entry indicates that it does not need to be updated does not need to be evaluated further.

4.3.5. Inform the responsible site program or process owner of the results of the review of their application and any changes arising from the above reviews. For example if the AOV ranking is updated, inform the AOV program manager of the changes and provide the revised results to them. The owner must informed even if no changes result from the model update.

4.3.6. Deferral of any of the above actions in the change management plan beyond a six month completion period should be approved by the Senior Manager Risk Management.

4.3.7. Ensure the updated model is stored and distributed in accordance with ER-AA-600-1014 (ref. 6.6).

4.4. Unscheduled Update Process

As noted in Section 4.1.3, the need for an unscheduled PRA update may arise because evaluation of a URE indicates that the risk significance of the PRA revision involved is such that it should not be delayed until the next scheduled periodic update.

4.4.1. The Model Owner will prepare a Project Plan for the unscheduled update (if the complexity of the unscheduled update warrants a Project Plan) which identifies tasks to be performed, a schedule to complete each task and resources to carry out the tasks.

4.4.2. Data Collection

OBTAIN AND REVIEW data in the same manner as described in Section 4.2.2 but limited in scope appropriate to the purpose and requirements of the unscheduled PRA update.

- **CONSIDER** only those data sources which are necessary to support the purpose and scope of the update
- **EVALUATE** open UREs to determine those that will be included in the update

In the performance of an unscheduled PRA update, it is not necessary to obtain the concurrence of the Risk Management Director for UREs deferred to the next periodic PRA update.

4.4.3. Data Screening and Analysis

The activities specified in Section 4.2.3 **NEED NOT BE DONE** unless they are necessary to support the scope of the unscheduled PRA update. The scope of those activities performed should be limited appropriate to the requirements of the unscheduled PRA update.

4.4.4. Model Changes

UPDATE the PRA technical elements appropriate to the requirements of the unscheduled PRA update. **CONSIDER** only those technical elements listed in Section 4.2.4 which are necessary to support these requirements. **CONSIDER** only the scope of these technical elements necessary to support the requirements of the unscheduled PRA update.

4.4.5. Quantification

QUANTIFY the model, **REVIEW** the model per Attachment 2 using the items appropriate to the changes incorporated and **PREPARE** a CDF and LERF documentation category 1 document for model changes made in the unscheduled PRA update that significantly change baseline CDF and/or LERF numbers, including importance measures. Significance is based on engineering judgment considering the criteria specified in the EPRI PSA Applications Guide (Ref. 6.1) for permanent changes.

1. **PREPARE** a documentation category 1 record of the updated PRA quantification, and **INCLUDE** the results of review of current applications which are significantly impacted by the updated PRA

   NOTE: Significance is based on whether the current application would no longer adequately represent the as-built, as-operated plant if not revised using the updated PRA. A qualitative review is acceptable if it clearly demonstrates that there is no significant impact on a current application.

2. **UPDATE** other PRA documentation appropriate to the scope and impact of the unscheduled PRA update. Normally this would be done as a revision of the documentation. This may also be done by **PREPARING** addenda to existing documentation category 1 documents or **PREPARING** other retrievable documentation. **UPDATE** the appropriate category 1 PRA documentation using the contents of these addenda or other documentation at the next periodic PRA update.

3. Consider the impact on or need to **UPDATE** other PRA models such as fire and seismic. If update is deferred create UREs specific to those models as necessary.

4.4.6. Approval

Approve per Section 4.2.6.

4.4.7. Documentation and Rollout

The activities specified in Section 4.3 **NEED NOT BE DONE** unless they are necessary because the updated PRA is to be used in application(s) which would make them appropriate. If necessary, within six months of approving the PRA model category 1 documentation:

1. If the updated PRA is to be applied to provide quantification results used in the on-line risk monitor, **UPDATE** Training Aids & Posters if the changes are significant, **PROVIDE** training as appropriate, **REVISE** the on-line risk monitor, **REVISE** the baseline CDF and LERF results used for trending, and **REVISE** any directly-associated applications

   NOTE: Directly associated applications are those in which the risk significance of on-line maintenance activities has been evaluated such as license amendments to provide extended allowed outage times.

2. If the updated PRA is to be used in any other current application, then **REVIEW** the impact to that application, and **SCHEDULE** revisions as appropriate

4.5. Ongoing Data Review

During the period between periodic updates, the SRME should review the following for changes that will impact the PRA model:

4.5.1. Plant Design Changes

Evaluate identified modifications for impact on the PRA model when a modification is initiated that may impact the PRA model and prepare a URE or unscheduled update to the PRA model as required.

4.5.2. Procedure Changes

REVIEW all procedures in HRA procedures list and new procedures quarterly for changes that can impact the PRA model. The Model Owner will maintain a list of HRA procedures to be reviewed. These procedures will include all procedures that are used in an HEP calculation (ex. emergency operating procedures) except for those tied to precursor events such as miscalibrations. Other procedures may be included in the list of procedures based on the Model Owner's judgment.

ISSUE a URE for procedure changes that are determined to have a possible impact on the PRA model. The Site Risk Management Engineer is expected to consult with personnel in the Training and Operations departments and participate when procedure changes that could have a significant impact to plant safety are considered.

4.5.3. Engineering Calculations

SCREEN revised and new Site Engineering Calculations on a quarterly basis. The Site Risk Management Engineer will **INITIATE** a URE if it is determined that a calculation may impact the PRA model and further evaluation and/or incorporation into the PRA model is required.

4.5.4. Document the above reviews in a manner that will allow a RME to determine whether a change has been reviewed for impact.

5. DOCUMENTATION

5.1. **PREPARE** PRA update documentation according to ER-AA-600-1012, "Risk Management Documentation."

6. REFERENCES

6.1. EPRI TR-105396, "PSA Applications Guide," August, 1995

6.2. ASME RA-S-2002 and addenda, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications."

6.3. IT-AA-101, "Digital Technology Systems (DTS) Quality Assurance Procedure."

6.4. ER-AA-600-1011, "Risk Management Program."

6.5. ER-AA-600-1012, "Risk Management Documentation."

6.6. ER-AA-600-1014, "Risk Management Configuration Control."

6.7. ER-AA-600-1047, MSPI Basis Document

7. ATTACHMENTS

7.1. Attachment 1: Sample Updating Requirements Evaluation Form

7.2. Attachment 2: Review of updated PRA Model

ATTACHMENT 1
Sample Updating Requirements Evaluation Form
Page 1 of 1

Exelon PSA URE

URE Number:		Station:		Unit		Initiated By:	
Reason	☐ Design Change ☐ Improvement	☐ Procedure / Policy ☐ Expansion	☐ Calculation ☐ Other			Date:	

URE Description:

Evaluation Performed By:		Date:	
Evaluation Notes:	Significance:	Category:	

PSA Action
- ○ None. The PSA model is not impacted by this UR
- ○ The PSA model is impacted and will be revised.

Schedule:
- ○ Immediate attention required for applications.
- ○ Next Periodic Update, consider in applications.
- ○ Other - see comments below.

PSA Action - Comments:

Model Revision Completed: ☐ **Document Revision Completed:** ☐

Additional Comments:	Resolution:

URE Completed: ☐	URE Resolved By:		Completion Date:	
	Reviewed By:		Reviewed Date:	

Updating Requirement Evaluation

ATTACHMENT 2
Review of updated PRA model
Page 1 of 2

The below reviews are of the base full power internal events model. They do not supplant reviews of individual portions of the update such as analysis files, notebook revisions or supporting calculation such as updated maintenance frequencies. These reviews represent the minimum required, additional reviews may be performed at the discretion of the RME.

1. For a revision to a fault tree's logic structure, review at a minimum the top 20% of the resultant cutsets for validity. If there is an opposite train or Unit fault tree available, compare the results against each other. For example compare the A train cutsets to the B train cutsets. This type of review is not required for other fault tree changes such as revision to gate or basic event descriptions or names.

2. Review the top 500-600 cutsets from the integrated results. Are the cutsets valid?

3. Review a sampling of non-dominant sequences/cutsets to determine that they are reasonable and make physical sense.

4. Verify that truncation limits result in convergence of results toward a stable value. This applies to all truncation limits used (for example both event tree and fault tree truncation values should be evaluated for convergence in a WinNUPRA model).

5. Review the top 100-150 cutsets against the previous model results. Are the differences explainable by the changes to the model? For example if a specific initiating event frequency dropped by 10%, the associated cutsets may drop out of the top 100. Also if credit for a system or function changes it may shift the results.

6. Review the results by initiator against the previous model. Are the difference in ranking and absolute value explainable?

7. Perform an initial draft ranking for at least one valve type. Review against previous results.

8. Perform an initial draft Maintenance Rule risk significance listing. Review against previous results.

9. Perform ORAM-Sentinel or PARAGON cases for selected high CDF singles and combinations, selected low CDF singles and combinations. Selected cases should include systems where significant work was performed in the update. For example pick electrical cases where work was done on the LOOP event trees. Review against previous cases.

ATTACHMENT 2
Review of updated PRA model
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10. For sites with separate unit models, compare the dominant sequence frequencies between the two models.

11. Document the above reviews and resolve any identified issues prior to final signoff of the PRA model for use.