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ComEd

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July 7, 2000

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

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: Response to 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"

References: (1) Letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Request for Amendment to Technical Specifications - Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators," dated January 20, 2000.

(2) Letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Supplement To Request for Amendment to Technical Specifications - Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators," dated April 3, 2000.

(3) Letter from G. F. Dick (U. S. NRC) to O. D. Kingsley, "Byron and Braidwood, Units 1 and 2 - Request for Additional Information Related to the Requested Extension of the Allowed Outage Times for the Emergency Diesel Generators," dated June 13, 2000.

A001

A License Amendment Request (LAR) for the Byron Station and the Braidwood Station was submitted to the NRC in Reference 1 and supplemented in Reference 2. The NRC subsequently issued a Request for Additional Information (RAI) letter in Reference 3. The RAI letter requested that additional information be provided within 30 days after receipt of the letter (i.e., by July 13, 2000). The requested additional information is provided in the attachments to this letter.

Should you have any questions concerning this letter, please contact Ms. Kelly M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Vice President - Regulatory Services

Attachments:
Attachment A: Response to Request for Additional Information Question #1
Attachment B: Response to Request for Additional Information Question #2

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

ATTACHMENT A

Braidwood Station, Units 1 and 2
Byron Station, Units 1 and 2

Response to Request for Additional Information Question #1

Reference: NRC letter dated June 13, 2000, "Byron and Braidwood Units 1 and 2 – Request for Additional Information Related to the Requested Extension of the Allowed Outage Time for the Emergency Diesel Generators"

Question #1

Please provide the details of any significant findings and observations from the WOG Peer Review Certification. Please include in the discussion any plant improvements or corrections that were made in the plant as a result of the findings.

Response to Question #1

The Peer Review Certification of the Braidwood Station Probabilistic Risk Assessment (PRA), also referred to as Probabilistic Safety Assessment (PSA), performed by the Westinghouse Owners Group (WOG) during the period of August 30, 1999, through September 3, 1999, resulted in seven Findings and Observations (F&O) with the significance level of "A" and 19 F&O with the significance level of "B." The significance levels of the WOG Peer Review Certification process have the following definitions.

- A - Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.
- B - Important and necessary to address, but may be deferred until the next PRA update.

To address and disposition each of the F&O, Commonwealth Edison (ComEd) Company used the Update Requirement Evaluation (URE) forms in the PRA update evaluation process, as defined by ComEd procedure NEP-17-04, "Nuclear Engineering PSA Model Update Procedure." The F&O with the significance levels of "A" and "B" were reviewed, dispositioned, and documented before the Emergency Diesel Generator (EDG) Allowed Outage Time (AOT) Extension License Amendment Request (LAR) was submitted. The following table provides a summary of the significance levels A and B F&O and the corresponding resolutions. The F&O and the resolutions to the F&O are applicable to both the Byron and Braidwood Stations, since both stations use the same PRA model. The designators for the Items of the F&O are as follows.

- IE - Initiating Event
- AS - Accident Sequence Analysis
- TH - Thermal Hydraulic Analysis
- SY - System Analysis
- DA - Data Analysis
- HR - Human Reliability Analysis
- DE - Dependency Analysis
- QU - Quantification
- MU - Maintenance & Update

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
IE-3	The DLSX (i.e., dual loss of Essential Service Water (i.e., SX)) initiating event frequency is dominated by common cause failure to run the SX pumps. The IE Notebook identifies a recovery factor of 0.98. There is no basis or documentation for this recovery factor.	A	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. The necessary technical basis for applying a 98% recovery factor has been documented and a URE was prepared to update the IE Notebook to incorporate this documentation.
IE-6	Failure of the Reactor Coolant Pump (RCP) seal Loss of Coolant Accident (LOCA) is not included as an initiating event. Small non-isolable pipe break LOCA is included in the analysis with a frequency of 3.76E-4/year (Electric Power Research Institute). This value is very close to the value of 5E-4/year provided in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995." NUREG/CR-5750 also provides a frequency of 2.5E-3/year for RCP seal failures.	B	Based on this F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this F&O are already reported in the LAR. The PRA model was changed by adding RCP Seal LOCA initiator (9.2E-4/year) to the non-isolable small LOCA initiator (3.76E-4/year). The initiating event frequency cited in NUREG/CR-5750 was adjusted from 2.5E-3/year to 9.2E-4/year based on guidance provided in NUREG/CR-6582, "Assessment of Pressurized Water Reactor Primary System Leaks," Chapter 15.
IE-8	The documentation of the support system (i.e., special) initiating events in the IE Notebook needs to be completed (i.e., Section 3.5.3, "Support System Events"). Loss of Instrument Air and Loss of Non-Essential Service Water System initiator descriptions are specific examples. The dual and single unit loss of Essential Service Water System descriptions should also be separated. It is unclear whether the recovery described applies to both DLSX and LSX (i.e., loss of SX).	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. A URE was generated to review the IE Notebook and all System Notebooks that contain a discussion on support system initiating events, and to update the IE Notebooks to include proper documentation on the support system initiating events. The issue dealing with recovery of DLSX and LSX will be clarified in accordance with the resolution provided for F&O IE-3.
AS-3	Event ALTFW-TRANS (i.e., alternate source of feedwater) is not considered when both the centrifugal charging (i.e., CV) pumps and safety injection (SI) pumps have failed. This appears to be inconsistent with the guidance in Functional Restoration (FR)	B	Omitting the ALTFW-TRANS is conservative and has a negligible impact of the results reported in the LAR since ALTFW is disabled upon a Loss of Offsite Power (LOOP) due to loss of power to the 4 kV non-Engineered Safety

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	Procedure FR-H.1, "Response to Loss of Secondary Heat Sink."		Feature (ESF) buses. If credit were taken for ALTFW when both the centrifugal charging pumps and SI pumps have failed, it would reduce the Core Damage Frequency (CDF). Its inclusion could be justified since Procedure FR-H.1, Step 14, instructs the Operators to continue attempts to establish Reactor Coolant System (RCS) feed flow if an RCS feed path can not be established, and then directs a return to Step 4 where Operators are instructed to establish alternate feedwater in steps 6 -11.
AS-6	For the first hour of Station Blackout (SBO), the probability of a 480 gpm RCP seal LOCA is assumed to be zero. This modeling is not consistent with the generally accepted current industry interpretation of the NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," model. The model in the NUREG-1150 analysis was not documented correctly. This is a known error in the NUREG.	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. The necessary technical basis for assuming zero probability of an RCP seal LOCA during the first hour has been documented and a URE was prepared to update the Event Tree Notebook to incorporate this technical basis and results of a sensitivity analysis to demonstrate that there is negligible impact.
AS-9	Credit is given for operation of the diesel driven Auxiliary Feedwater (AFW) pump past the time when the batteries would become completely discharged in a SBO event. The design battery life is four hours, although the batteries could probably last much longer, after which instrumentation necessary to properly control AFW flow would be no longer available. This condition should be investigated and described in a more complete fashion.	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. The necessary technical basis for operation of the diesel driven AFW pump has been strengthened and a URE was prepared to update the Human Reliability Analysis (HRA) Notebook to incorporate this documentation.
TH-2	The basis for stating that equipment will survive for 24 hours in the Equipment Survivability Notebook is in some cases engineering judgment and in at least one case reviewed (e.g., SI pumps) based on bearing temperature calculations. The conclusion does not appear to be supportable. That is, the calculated maximum allowable bearing temperature is 169°F,	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. The SI pump success criterion was based on empirical data that showed that this type of pump began to show signs of bearing distress at 250°F. A conservative heat

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	and the statement is made that by engineering judgment the pump will survive for 24 hours at this temperature.		transfer analysis showed that a 250°F bearing temperature corresponds to a 169°F ambient room temperature. Furthermore, the results reported in the LAR are insensitive to this success criterion because room heat-up analyses show that following a loss of room cooling in the SI pump cubicle, the temperature would rise at < 1°F per hour. Assuming an initial temperature of 108°F, the room temperature would reach only 152°F in 24 hours, which provides ample time to provide an alternate means of room cooling.
TH-3	It is difficult to match success criteria to specific analyses and fault tree cases. Success criteria are stated and explained in the Success Criteria Notebook, but there is no cross-reference to how these criteria correspond to event sequence paths.	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. A URE was prepared to update the Success Criteria Notebook to incorporate this documentation.
TH-5	The discussion of AFW success criteria in the Success Criteria Notebook section on LOOP does not match the modeling of LOOP in the Event Tree Analysis Notebook.	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. The PRA model used for the LAR is correct. A URE was written to resolve the inconsistencies in the documentation in order to properly state the fault tree top logic.
SY-1	The SX cross-tie is credited for EDG recovery (i.e., both EDGs can be supplied by either SX header). The EDG auto-start blocks most trips, including jacket water temperature. Without procedural guidance, when would the operators turn off the EDGs without cooling water?	A	Based on this item of the F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this F&O are already reported in the LAR. The system fault tree was modified to disable the SX cross-tie across the units for EDG jacket cooling water during a LOOP event. A URE was written to update EDG and SX System Notebooks accordingly.
SY-4	The effects of test misalignment are, in general, not modeled at the system level. The review of station surveillance procedures	B	Analysis for pre-initiators was conducted as part of our response to the Byron and Braidwood Station Individual

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	describes the effects of the testing, and if the system is taken out of its normal alignment, unavailability is included in the models. However, operator error of failure to restore is not included in the system models.		Plant Evaluation (IPE) Request for Additional Information (RAI) on March 27, 1997. The RAI states that to identify potential pre-initiator vulnerabilities, a search was conducted of records from 1992 through 1995 to identify events that had occurred at either Byron or Braidwood Station that included personnel error, out-of-service error, testing error, or miscalibration as a cause. Based on this screening analysis, most pre-initiators were screened out from the IPE model due to their low probability values. These low values are based on there being independent verification steps for system restorations in the relevant test procedures. The current PRA system model treats these pre-initiators in the same way. Because of their low probabilities, these pre-initiators will have an insignificant impact on the base PRA and LAR results. The pre-initiator analysis will be revised during next PRA update, and its scope will consider the findings and observations identified in this item of the F&O as well as in F&O Item HR-3 and F&O Item DE-7.
SY-6	The fault tree logic for automatic start of the motor-driven AFW pump (MDP) assumes that a SI signal is always initiated. This is probably not true.	B	Based on this item F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this F&O are already reported in the LAR. The PRA model was changed so that SI signal does not auto-start the motor-driven AFW pump. A URE was written to update the AFW System Notebook.
SY-8	Fault tree AFW top logic in the System Notebook is obsolete (i.e., March version). The Success Criteria Notebook indicates that this specification is based on bounding the requirements for normal transient with the requirements for response to loss of	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. A URE was written to resolve the inconsistencies in the

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	heat sink (i.e., Procedure FR-H.1).		documentation and fault tree top logic.
DA-1	▪ LOOP Given Transient: Not modeled. Discuss justification for completeness. ▪ RCP Seal Failures: Uses NUREG-4550, "Analysis of Core Damage Frequency: Internal Events Methodology," model; well documented. Does not address 480 gpm RCS seal LOCA scenario during the first hour. Uses computer code Modularized Accident Analysis Program (MAAP) to estimate and use T1200. Need to justify the exclusion of this LOCA scenario in MAAP.	B	Double initiators are not considered to be credible in the PRA model used for the LAR. In addition, LOOP Initiating Event Frequency (IEF) includes all plant-centered events, including the events both before and after the plant trips. A sensitivity analysis of a simultaneous General Transient and a LOOP initiator shows that the base CDF increases by only about three percent. A similar sensitivity analysis with the Braidwood EDG 1A out-of-service showed a slightly smaller increase in CDF. Thus, there is actually a very small decrease in delta CDF (i.e., about 1E-08/year). For the RCP Seal Failure portion of F&O Item DA-1, see the resolution provided for F&O AS-6.
DA-5	Asymmetric common cause groups were defined for the EDGs. The justification is presented as "major diesel overhauls are performed during a refueling. After the overhaul, the affected diesels are different from those of the other unit." The reviewers do not agree with this justification.	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. Justification for using the asymmetric Common Cause Failure (CCF) group for the EDGs is as follows: The quantification of the CCF event probability already includes 3/4 CCF groups in the 4/4 CCF group by setting the delta parameter to 1.0. This is a conservative treatment as compared to explicitly modeling the 3/4 CCF groups separately. In addition, all necessary Multiple Greek Letters (MGL) parameters for the asymmetric CCF group were considered in the 2/4 CCF group.
HR-2	The ComEd PRA Guideline (i.e., Red Book) states that Cause Based Decision Tree (CBDT) method used in the analysis builds upon the method described in the supporting EPRI document. Extra branch points have been added to the	B	This is a documentation enhancement issue and has no impact on the PRA model used for the LAR. A detailed explanation of the adjustment to the EPRI

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	standard CBDT decision trees and additional decision trees have been included. Thus, in effect, a new post-initiator analytical method has been used.		standard CBDT has been written, and is being incorporated into the HRA Notebook.
HR-3	Only selected pre-initiator human errors were included in the model. Post-maintenance errors were excluded on the basis of a review of operating history which did not reveal any vulnerabilities. Post-test misalignment errors were only included for those valves whose position is not indicated in the main control room. Miscalibration errors were excluded. The exclusion of common cause miscalibration errors from the model is not consistent with industry practice.	B	Miscalibration errors were investigated in the previous analysis performed in response to the Byron and Braidwood Station IPE RAI, and these events were screened out based on the low probability values assessed. Investigated instruments included instrumentation used to initiate automatic actions and instruments used by the Operators in responding to an initiating event. The common cause events related to miscalibration errors were not included in the PRA model used for the LAR. The potential for multiple instrument miscalibration is minimized by the practice that multiple channels of plant instrumentation used to measure a parameter are not worked on in the same day by the same instrument technician using the same test equipment. Additionally, calibration results are reviewed to compare the recorded "as found" value with the "as left" value to identify discrepancies. Similarly, for parameters with more than one channel, miscalibration would be identified at the end of the calibration activity by a simple "channel check," i.e., comparing the meter reading of the just calibrated channel to an adjacent meter indicating the same parameter. As stated in the response to F&O Item SY-4, the screened out pre-initiators due to low probability values are expected to have an insignificant impact on the results of PRA and LAR. Additionally, a potential for common cause miscalibration errors is judged to be minimal at Byron and Braidwood Stations due the practice described above.

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
			However, a URE was written to further investigate applicability of common cause miscalibration errors during next PRA update.
HR-4	Steam Generator Tube Rupture (SGTR) mitigation following loss of AFW presumes that Operators will enter procedure FR-H.1 and follow the normal procedure for attempting to start Alternate Feedwater/Initiating Bleed and Feed when level reaches 27% Wide Range. In fact, under SGTR conditions, entry into Procedure FR-H.1 will not be straight forward, as the level in the ruptured Steam Generator (SG) will be above the prerequisite 10% unless Operators open the SG Power Operated Relief Valves (PORVs). Neither the event tree structure nor the HRA analysis reflects this path. Currently there are no success criteria to validate the use of bleed and feed under these conditions.	B	Based on this Item of the F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this Item of the F&O are already reported in the LAR. A URE was written to include the documentation of the Human Error Probability (HEP) revisions in the HRA Notebook, and to revise the Event Tree Analysis Notebook and Success Criteria Notebook.
HR-5	It is understood that neither Operator input or simulator experience has been input into the development of HEPs included in the HRA Notebook, Revision 2, Supplement 1, as yet. This may have a significant impact on time critical actions modeled using Human Cognitive Reliability (HCR) approach. In particular, Bleed and Feed Operator actions which are included in dominant sequences should be reviewed.	A	The most risk significant actions and those that are important for evaluating the changes in risk due to the EDG AOT Extension were addressed as suggested by the Certification Team. A number of risk significant actions in the PRA used for the LAR are quantified using conservative screening values in comparison with what would be expected from a more rigorous assessment. The fact that Operator interviews were not taken into account was compensated for by the conservative manner in which many other actions were quantified. Since the certification, the HRA has been validated using interviews of Operator trainers. This information resulted in a systematic review of 60 different Operator actions included in the PRA model. Of these, 10 actions were re-evaluated using new training insights. All 10 HEP

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
			values associated with these actions showed a reduction in the range of 20%-60%.
HR-6	Operator action for switchover to high pressure (HP) recirculation assumes the time window for action is from Lo-Lo Refueling Water Storage Tank (RWST) level to core damage (i.e., approximately three hours). In fact, if action is not taken much earlier to turn off the CV and SI pumps as directed by the procedure prior to the RWST reaching the point at which cavitation occurs, HP recirculation will not be successful. Time window for action is probably more like 20 minutes.	A	Based on this Item of the F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this Item of the F&O are already reported in the LAR. A URE was written to include the documentation of this HEP revision in the HRA Notebook.
HR-11	The Feed and Bleed Analysis has been treated as a time critical action in the PRA and has been evaluated using the Human Cognitive Reliability (HCR) model. To implement this method some key timing information is required. In particular, the following information is needed. ▪ The time interval between the cue (i.e., plant symptom recognized in the Emergency Operating Procedure) and the last possible moment the Operator can take the action before the undesirable state occurs (i.e., in most cases core damage). ▪ The time required for the Operator to implement the action once he has decided to do so. ▪ T1/2, the median time required by crews to take the action after the cue has occurred. The current timings used in the Bleed and Feed Analysis do not appear to be substantiated, and the values are assigned inconsistently.	A	Based on this Item of the F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this Item of the F&O are already reported in the LAR. A URE was written to include the documentation of these HEP revisions in the HRA Notebook.
DE-6	The internal flooding analysis workbook identifies the circulating water piping that is located in the Condensate Pump Pit (i.e.,	B	This Item of the F&O has no impact on the EDG AOT Extension calculation because flooding was not included

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	Walkdown notes for Turbine Building Basement Condensate Pump Pit, Section 8.2-1 second page last paragraph). The internal flooding analysis, section 7.2, does not include the effects of failure of the circulating water piping although failure of the condensate piping is quantified. Because the circulating water piping includes expansion joints, the likelihood of a flood from this source could be higher than the flood scenario frequency that was developed for this area.		in the attached PRA model used for the LAR. Its exclusion is justified as stated in the attached response to RAI Question #2. Response to this Item of the F&O will be included in the revised flooding analysis to be performed following completion of the modifications described in the LAR.
DE-7	Common cause miscalibration of similar sensors does not appear to have been performed. Examples of locations where an analysis should have been performed include: AFW pump suction pressure trip sensors; Reactor Protection System (RPS) trip channels (i.e., pressurizer pressure, etc.)	B	The cutsets and importance list from Braidwood Station 1A EDG sensitivity case for EDG AOT Extension calculation demonstrate that common cause miscalibration of the AFW pump suction pressure trip sensors and RPS trip channels have negligible impact on the EDG AOT Extension calculation. Based on the results from two examples above, and the fact that most pre-initiators related to the miscalibration errors were screened out due to low probability, values are expected to have an insignificant impact on the results of PRA and LAR. As stated in the response to F&O Item HR-3, a potential for common cause miscalibration of similar sensors is judged to be minimal due to the station practice regarding instrument calibration. However, a URE was written to further investigate common cause miscalibration of similar sensors during next PRA update. See additional notes provided in the resolution of F&O Items SY-4 and HR-3.
QU-2	The absolute value of some of the cutsets may not be appropriate based on issues identified in other parts of the review. In particular the issue associated with the quantification of the Bleed and Feed HEP following loss of DC bus 111 may impact some sequence frequencies. The small break LOCA contribution may also change due to the increase in the LOCA	A	Based on this Item of the F&O, the PRA model was changed before the EDG AOT Extension calculations were performed. Thus, any impact on the results from this Item of the F&O are already reported in the LAR. Details are provided in response to other specific

SUMMARY OF WOG PEER REVIEW FINDINGS AND OBSERVATIONS			
F&O	**Observations by Peer Review Team**	**Significance**	**Plant Response or Resolution**
	IE frequency and HEPs associated with failure to go to recirculation and failure to stop the Resident Heat Removal (RHR) pumps.		certification F&O Items.
QU-7	At present only a parametric uncertainty analysis has been performed, although the Uncertainty Analysis Notebook clearly identifies and describes the requirement for addressing other types plant specific uncertainty issues. For example, cases where thermal hydraulic analyses predict only small margins for success in terms of the number of trains required, or the time available for operator actions, are prime candidates. Risk achievement analyses may be used to focus the search for potentially significant cases.	A	The Certification team reviewed an early draft of the Uncertainty Analysis Notebook. The uncertainty analysis has now been completed and documentation is in progress. The uncertainty analysis results show that Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," criteria for the baseline CDF (< 1E-4/year.) and Large Early Release Frequency (LERF) (< 1E-5/year.) have been met with a high degree of confidence. The distribution means for CDF and LERF are within 2% and 4% of the point estimates used in the LAR, respectively. An uncertainty analysis was also performed for the delta CDF, delta LERF, ICCDP (i.e., Incremental Conditional Core Damage Probability), and ILERP (i.e., Incremental Large Early Release Probability) risk metrics reported in the LAR. These also show that the respective decision criteria are met with a high degree of confidence.
MU-2	PRA Update Procedure NEP-17-04 addresses the items noted in sub-element MU-4. Per discussion with the Braidwood Station PRA engineer, there are plans to include future tracking of changes to the accident management program as well. He also indicated that he reviews industry studies for potential impact. This procedure has not yet been implemented at the Braidwood Station. As a result, it was not possible to determine, at the time of this review, the extent to which certain sub-elements are addressed in the Braidwood Station PRA program.	B	This Item of the F&O has no impact on the EDG AOT Extension calculation. This is neither a PRA model, nor a documentation issue. The implementation of NEP 17-04 is currently in progress at the Byron and Braidwood Stations. Full implementation is expected by December 31, 2000.

ATTACHMENT B

Response to Request for Additional Information Question #2

Braidwood Station, Units 1 and 2
Byron Station, Units 1 and 2

Reference: NRC letter dated June 13, 2000, "Byron and Braidwood Units 1 and 2 – Request for additional information related to the requested extension of the allowed outage time for the Emergency Diesel Generators"

Question #2

In Attachment E, Page E-11, of the submittal, it is stated that "at the completion of the flooding modification, and prior to implementation of the proposed changes, an updated internal flooding evaluation will be performed to confirm that there is no impact on the conclusions of the current risk evaluation of the proposed extended completion times." During the phone conference on June 5, 2000, the licensee stated that they do not anticipate any changes in the risk evaluation because the accident sequences relevant to the proposed AOT extension would not be impacted by the plant modifications. Please confirm this statement and provide a brief discussion to support this statement.

Response to Question #2

The Probabilistic Risk Assessment (PRA) model used as the basis for the License Amendment Request (LAR) to extend the Allowed Outage Time (AOT) for the Emergency Diesel Generators (EDGs) excluded representation of both flooding and the modifications which are described in the letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Request for Amendment to Technical Specifications - Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators," dated January 20, 2000, and as supplemented in letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Supplement To Request for Amendment to Technical Specifications - Extension of Allowable Completion Times and Surveillance Requirement Change for Emergency Diesel Generators," dated April 3, 2000. Neither flooding nor the proposed modifications have any influence on the Loss of Offsite Power (LOOP) scenerios, which lead to dependence on the EDGs. This is demonstrated by the results of the PRA sensitivity studies reported in the LAR. Since flooding equally impacts the base risk and the elevated risk due to an EDG being unavailable, the conclusion of the sensitivity studies relative to delta Core Damage Frequency/Large Early Release Frequency (Δ CDF/LERF) and Incremental Conditional Core Damage Probability/Large Early Release Probability (ICCDP/LERP) is unaffected. Additional sensitivity studies will be conducted in support of planning the proposed modifications, which will confirm that PRA scenarios involving flooding and the effects of the proposed modifications do not change the conclusions reported in the LAR. The PRA model used in support of the LAR is sufficient for demonstrating that the Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," acceptance criteria are met. Therefore, there will be no impact on the conclusions of the current risk evaluation of the proposed EDG AOT extension.