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

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Response to Request for Additional Information
License Amendment Request Regarding Relocation of Selected Technical Specification Surveillance Frequencies to a Licensee Controlled Document

References:

1. Letter from P. B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated October 30, 2009.

2. Letter from G. Edward Miller, U.S. Nuclear Regulatory Commission, to Charles G. Pardee, Exelon Generation Company, LLC, "Oyster Creek Nuclear Generating Station - License Amendment Request Regarding Relocation of Selected Technical Specification Surveillance Frequencies to a Licensee Controlled Document (TAC No. ME2494)," dated March 29, 2010.

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS). The proposed amendment would modify OCNGS TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled document. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. Draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On March 18, 2010, a teleconference was held between the NRC and Exelon to further discuss the additional information requested by the NRC. In

Reference 2, the NRC formally issued the request for additional information. The attachment to this letter provides a restatement of the questions along with Exelon's responses.

Exelon has concluded that the information provided in this response does not impact the conclusions provided in the original submittal (Reference 1).

This response to the request for additional information contains no regulatory commitments.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of April 2010.

Respectfully,

gbx Pamela B Cowan

Pamela B. Cowan
Director, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information
Attachment 2: Revised TS Page 4.2-1 Mark-up

cc:	Regional Administrator - NRC Region I	w/attachments
	NRC Senior Resident Inspector – OCNGS	"
	NRC Project Manager, NRR – OCNGS	"
	Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection	"
	Mayor of Lacey Township, Forked River, New Jersey	"

ATTACHMENT 1

License Amendment Request

Oyster Creek Nuclear Generating Station
Docket No. 50-219

License Amendment Request Regarding
Relocation of Selected Technical Specification
Surveillance Frequencies to a Licensee Controlled Document

Response to Request for Additional Information

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST REGARDING RELOCATION
OF SELECTED TECHNICAL SPECIFICATION SURVEILLANCE
FREQUENCIES TO A LICENSEE CONTROLLED DOCUMENT**

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS). The proposed amendment would modify OCNGS TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled document. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. Draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On March 18, 2010, a teleconference was held between the NRC and Exelon to further discuss the additional information requested by the NRC. In Reference 2, the NRC formally issued the request for additional information (RAI). The questions are restated below along with Exelon's responses.

RAI-01

The license amendment request (LAR) proposes to modify Surveillance Requirement (SR) 4.2.C.3 including the removal of the phrase "…for at least 20 control rods…" The modification or deletion of the number of control rods to be tested during performance of the SR was not considered as a part of the model TS change or in the model Safety Evaluation. Provide a justification for why allowing relocation, to the Surveillance Frequency Control Program (SFCP), of the number of control rods to be tested is acceptable. Alternatively, the proposed TS pages may be revised to retain the subject phrase.

RESPONSE

After further review and evaluation of TS 4.2.C.3, Exelon has concluded that the specific number of control rods to be tested is an integral part of the surveillance requirement itself. Therefore, Exelon proposes to leave the specific number of control rods to be tested in TS 4.2.C.3 and only relocate the number of days of cumulative power operation at which the surveillance must be conducted. Attachment 2 provides a revised TS page 4.2-1 mark-up to reflect this change.

RAI-02

Attachment 2 of the LAR, Table 2-1, Gap #1 justifies mission times exceeding 24 hours for loss of decay heat removal sequences as "current approach is judged to be reasonable for long term scenarios…" Given that, for assessing potential risk impact, assuming longer mission times generally increase conservatism by increasing the assumed failure probability of components, provide additional justification as to why this is an acceptable conclusion.

RESPONSE

The mission time used in the PRA for equipment run times is 24 hours unless explicit exception is noted to use a shorter mission time (e.g., batteries). Extending the Fail to Run (FTR) mission time beyond 24 hours for loss of Decay Heat Removal (DHR) sequences is considered to be an unnecessary complication and does not affect PRA insights and does not significantly affect the quantitative evaluation. Nevertheless, the thermal-hydraulic calculations used to support loss of DHR sequence success criteria extend beyond 24 hours to allow proper characterization of those sequences that have yet to reach a safe, stable state (e.g., time to core damage is greater than 24 hours).

The associated Supporting Requirement (SR) (SC-A5) [Reference 3] was identified as meeting Capability Category II/III by the 2006 OCNGS PRA Peer Review team. There were no Facts and Observations (F&Os) assigned for this SR. This SR was identified as a gap based on a Self-Assessment following the OCNGS 2008A PRA update. In addition, the second paragraph of the gap description in Table 2-1 of Attachment 2 of the LAR is a description of our current status. Therefore, the way that the gap is written may introduce confusion. It is not the intent of the PRA model to increase model conservatism; rather, the intent of the model is to provide as realistic an assessment of the risk as possible.

This is a documentation related issue and does not impact the quantitative analysis.

RAI-03

Attachment 2 of the LAR, Table 2-1, Gap #3 and Gap #4 justify the current component failure modes only by stating it is "…judged to include proper treatment…" Provide additional justification to support this conclusion.

RESPONSE

The associated SRs (SY-A12 and SY-A14) [Reference 3] were both identified as meeting Capability Category Met (All) by the 2006 OCNGS PRA Peer Review team. There were no F&Os assigned for these SRs. These gaps were identified based on a Self-Assessment following the OCNGS 2008A PRA update. This is a documentation related issue.

The PRA models the appropriate components and failure modes. For example, the PRA explicitly models active failure modes such as Failure to Start, Failure to Run, Failure to Open, and Failure to Close. Certain low probability component failure modes (e.g., Failure to Remain Open, Failure to Remain Closed, Plugging, or other passive failure modes) may not always be explicitly modeled if the failure mode represents less than 1% of the total component failure probability as stated in SY-A14. Passive failure modes (e.g., Bus Circuit Breaker Fails to Remain Closed) are explicitly modeled as judged necessary to model appropriate failure modes and dependencies.

The intent of the ASME PRA Standard SRs SY-A12 and SY-A14 are met. The documentation of the screening of these very low probability failure modes does not affect the PRA results, but the documentation of this screening has been identified by Exelon as an item to improve in the future.

The failure modes that would be examined as part of a change to a surveillance frequency under TSTF-425 would be active component failures, which are the same failure modes that are included in the PRA. Therefore, the model is at the appropriate level of detail for this application.

RAI-04

Attachment 2 of the LAR, Table 2-1, Gap #8 identifies failure to consider the quality of the plant procedures, administrative controls, and human-machine interface for both pre- and post-initiator human actions. The current status discusses a "potential upgrade" for pre-initiator actions. There is no status provided for post-initiator actions, although this gap is identified as "OPEN," and not partially resolved. Provide the specific status of this item for both pre- and post-initiator human actions. Provide an assessment of the significance of the deficiency for this application.

RESPONSE

The associated SR (HR-D3) [Reference 3] was identified as meeting Capability Category II/III by the 2006 OCNGS PRA Peer Review team. There were no F&Os assigned for this SR. This gap was identified based on a Self-Assessment following the OCNGS 2008A PRA update. This is a documentation related issue.

Subsequent to the original license amendment request submittal, an assessment of the OCNGS procedures, administrative controls, and the human-machine interface was performed for both pre-initiator and post-initiator human actions. The results indicated that OCNGS procedures, administrative controls, and the human-machine interface are equal to or slightly better than many other boiling water reactors that have been reviewed.

The Human Reliability Analysis (HRA) development for the PRA assumed that the OCNGS procedures, administrative controls, and the human-machine interface for Human Error Probability (HEP) evaluation were consistent with industry norms. No benefit from higher quality procedures or administrative controls was included. Therefore, the assumptions made in the HRA development are appropriate and require no quantitative changes.

The Gap #8 critique is primarily aimed at documenting the basis for the assessment of the OCNGS written procedures and administrative controls.

No changes to the post-initiator HEPs or the pre-initiator HEPs resulted from this review of the procedures, administrative controls, and human-machine interface.

RAI-05

Attachment 2 of the LAR, Table 2-1, Gap #9, Gap #10, Gap #11, and Gap #12 justify that estimating relevant plant data (i.e., demands and standby time) rather than determining them from actual plant data is sufficient. Estimating data is consistent with Capability Category I only. Justify why estimating this data is an acceptable approach to resolving the gaps.

RESPONSE

A summary of the Capability Categories assigned by the OCNGS Peer Review team in 2006 for the subject Gaps are as follows:

Gap	SR	CC	F&O
#9	DA-C6	Met (All)	None
#10	DA-C7	I	Level “C”
#11	DA-C8	I	Level “C”
#12	DA-C10	II	Level “C”

SRs DA-C6 and DA-C10 [Reference 3] were identified as meeting Capability Category II as part of the peer review. However, these two SRs were identified as gaps based on a Self-Assessment following the OCNGS 2008A PRA update. For SRs DA-C7 and DA-C8 [Reference 3], both the OCNGS peer review and the Self-Assessment identified these SRs as potential areas of enhancement.

The failures of equipment (numerator of the failure rate estimation) are recorded from OCNGS operating experience. The number of failures is a critical input to the failure rate evaluation and these failures are based on data.

Consistent with SR DA-C7, the number of plant specific demands is based on the number of surveillance tests which is also consistent with actual practice. The number of additional planned and unplanned maintenance events is considered to be a small addition to the number of demands based on surveillance tests. This may lead to a slight conservatism.

The demands are the denominator of the failure rate calculation. For standby systems, nearly all of the demands are surveillance tests. Therefore, basing the number of demands on the surveillance test frequencies is judged consistent with actual experience, but may not account for some plant operational demands. Therefore, the use of scheduled and prescribed surveillance tests as the denominator would be equal to or less than the actual number of demands. This approach captures the appropriate number of demands performed during each test (i.e., reflects the number of demands within the test procedure if more than one). Unplanned tests are not included in the number of demands. This may then lead to a slightly higher failure rate than might be calculated from the use of the actual demand count. This difference is judged to not significantly impact the risk profile. The use of the frequency of OCNGS component surveillance tests is judged to be appropriate because the surveillance tests are monitored under strict plant and corporate procedures.

Additionally, the NEI 04-10 methodology, Step 8, requires that an appropriate time-related failure contribution be utilized in the surveillance frequency change assessment, and Step 14 requires that sensitivity studies regarding the choice of that value be performed.

<u>RAI-06</u>

Attachment 2 of the LAR, Table 2-1, Gap #10 states that the system engineer estimate of demand data is adequate for the probabilistic risk assessment, however, Gap #13 states that system engineer experience level is inadequate to obtain insights on maintenance unavailability. Describe how, as described in the LAR, system engineer experience is adequate to resolve Gap #10 but not Gap #13. Alternately, provide another justification for the resolution of Gap #10 that does not utilize system engineer experience.

<u>RESPONSE</u>

For Gap #10, the number of demands and standby time can be determined by the System Engineers because they are extremely knowledgeable on the various surveillance tests and the prescriptive schedules involving their systems. Therefore, the System Engineer determination of demand data is based on the surveillance test procedures which are judged to be appropriate for input to the PRA.

For Gap #13, SR DA-C12 [Reference 3] was identified as meeting Capability Category II/III by the 2006 OCNGS PRA Peer Review team. There was a Level "C" F&O assigned for this SR related to documenting additional input from plant maintenance and plant operations for maintenance unavailability estimates. This gap was identified based on a Self-Assessment following the OCNGS 2008A PRA update. This is a documentation related issue.

The corresponding SR (DA-C12) identifies the following to meet Capability Category II/III:

> **INTERVIEW the plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events.**

Refer to the Table 2-1 excerpt included at the end of this attachment for updated text associated with Gap #13. The text for "Description of Gap" has been revised to state that the System Engineers are knowledgeable with respect to the maintenance unavailability times as input to the baseline maintenance unavailability probabilities. The intent of the gap is to enhance the PRA documentation to note that the System Engineer input is not explicitly used to support the development of Error Factors for maintenance unavailability probabilities as input to the parametric uncertainty analysis.

<u>RAI-07</u>

Attachment 2 of the LAR, Table 2-1, Gap #14 identifies a deficiency in the flood area definitions which, if corrected, might introduce new flood initiators, but further states that significant flood scenarios and propagation paths are already appropriately modeled. Provide the basis for this conclusion regarding significant flood scenarios. Further describe any conclusions regarding the impact of this deficiency documented by the August 2008 focused scope peer review for internal flooding.

<u>RESPONSE</u>

The associated SR IF-A1a [Reference 3] was identified as meeting Capability Category II/III by the 2008 OCNGS Focused PRA Peer Review team on the Internal Flooding element. There was a “Suggestion” level F&O assigned for this SR. This gap was identified based on a Self-Assessment following the OCNGS 2008A PRA update. This is a documentation related issue.

The referenced SR is as follows for Capability Category II:

> DEFINE flood areas at the level of **individual rooms or combined rooms/halls for which plant design features exist to restrict flooding.**

The plant areas are divided in a logical method to highlight those sources and targets, including the propagation paths.

Information from the August 2008 focused scope peer review for internal flooding regarding the associated F&O is as follows:

- Fact and Observation Number – IF-A1-01

- Discussion of Issue – *"Table 3-1 of the Internal Flood Walkdown Report identifies the fire zones that are included in each flood area with each flood area typically containing multiple fire zones. However, the basis for identifying each of the flood areas as independent and the basis for including multiple fire zones into one flood area is not described.”*

- Basis for Significance – *“The physical boundaries of each fire zone are described on an area-by-area basis in Section 7 of the FHA. The specific boundaries included can be determined on a zone-by-zone basis by performing a detailed review of the FHA. A spot check of flood area TB-FA-11 was performed and the bases for combining the multiple fire zones into the single flood area appeared to be reasonable.”*

- Possible Resolution – *“Add a description of the physical boundaries used to define the flood areas to the Internal Flood Notebook.”*

The peer review team identified that a check of flood area TB-FA-11 did not identify any concerns related to the definition of flood areas and their potential impact on the quantitative results. The identified resolution was to enhance the documentation for defining the flood areas.

The OCNGS internal flood analysis groups multiple flood areas together if the propagation timing and consequences are identified to be sufficiently similar, although not identical. For example, the OCNGS internal flood analysis models a single initiating event for any Feedwater/Condensate pipe flooding event originating on a specific elevation of the Turbine Building because any Feedwater/Condensate flooding on that elevation results in essentially the same flooding propagation path (flood Turbine Building basement) and consequence (fail Feedwater/Condensate and deplete the condensate storage tank).

The OCNGS interpretation of this SR is that flood sources that may propagate to the same targets within approximately the same time frame can be considered to be the same “flood area.” This interpretation has resulted in 39 internal flood initiators for the OCNGS PRA.

Based on the above information, the OCNGS flood areas are judged to be appropriately defined as indicated by the peer review results and capture the significant internal flood initiating events. The documentation enhancement will clarify that the existing flood boundaries are correct and adequate.

REFERENCES:

1. Letter from P. B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated October 30, 2009.

2. Letter from G. Edward Miller, U.S. Nuclear Regulatory Commission, to Charles G. Pardee, Exelon Generation Company, LLC, "Oyster Creek Nuclear Generating Station - License Amendment Request Regarding Relocation of Selected Technical Specification Surveillance Frequencies to a Licensee Controlled Document (TAC No. ME2494)," dated March 29, 2010.

3. Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-Sb-2005, December 2005.

Table 2-1
Status of Identified Gaps to Capability Category II
of the ASME PRA Standard(1)

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to Application
Gap #13	SR DA-C12 was identified as meeting Capability Category II/III by the 2006 OCNGS PRA Peer Review team. This Gap was identified based on a Self-Assessment following the OCNGS 2008A PRA update. This is a documentation related issue. It is noted that the System Engineers are knowledgeable with respect to system unavailability times and the potential ranges of system unavailability times as input to the baseline mean system unavailability probabilities. However, no interviews of plant staff were performed to generate uncertainty estimates of unavailability per maintenance act (e.g., develoment of Error Factors to support the parametric uncertainty analyses). Enhance the PRA documentation to note that the System Engineer input is not explicitly used to support the development of Error Factors for maintenance unavailability probabilities as input to the parametric uncertainty analysis.	DA-C12	Open. This deviation from the SR is not considered to significantly alter the PRA qualitative or quantitative results becase it only influences the determination of the error factors assigned to maintenance unavailabilities.	Not significant because the model is consistent with data from the plant MR database, so there will not be an impact on unavailability hours used in the model.

(1) This table is an excerpt from Table 2-1 of the LAR with text modified to respond to NRC RAI-06.

ATTACHMENT 2

License Amendment Request

Oyster Creek Nuclear Generating Station
Docket No. 50-219

License Amendment Request Regarding
Relocation of Selected Technical Specification
Surveillance Frequencies to a Licensee Controlled Document

Revised TS Page 4.2-1 Mark-up

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. SDM shall be verified:
   1. Prior to each CORE ALTERATION, and
   2. Once within 4 hours following the first criticality following any CORE ALTERATION.

B. The control rod drive housing support system shall be inspected after reassembly.

C. The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

   1. For all control rods prior to THERMAL POWER exceeding 40% power with reactor coolant pressure greater than 800 psig, following core alterations or after a reactor shutdown that is greater than 120 days.

   2. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either “a” or “b” as follows:

      a.1 Specifically affected individual control rods shall be scram time tested with the reactor depressurized and the scram insertion time from the fully withdrawn position to 90% insertion shall not exceed 2.2 seconds, and

      a.2 Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure prior to exceeding 40% power.

      b. Specifically affected individual control rods shall be scram time tested at greater than 800 psig reactor coolant pressure.

   3. ~~On a frequency of less than or equal to once per 180 days of cumulative power operation~~**In accordance with the Surveillance Frequency Control Program**, for at least 20 control rods, on a rotating basis, with reactor coolant pressure greater than 800 psig.

D. Each partially or fully withdrawn control rod shall be exercised ~~at least once per 31 days~~**in accordance with the Surveillance Frequency Control Program**. This test shall be performed within 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.