Exelon Generation.

RS-14-299
RA-14-083

10 CFR 50.54(f)

December 19, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike,
Rockville. MD 20852

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

Subject: Exelon Generation Company, LLC Expedited Seismic Evaluation Process Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident

References:

1. NRC Letter, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 12, 2012 (ML12053A340)
2. NEI Letter, Proposed Path Forward for NTTF Recommendation 2.1: Seismic Re-evaluations, dated April 9, 2013 (ML13101A379)
3. Seismic Evaluation Guidance: "Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 – Seismic", EPRI, Palo Alto, CA: May 2013. 3002000704 (ML13102A142)
4. NRC Letter, Electric Power Research Institute Report 3002000704, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations, dated May 7, 2013 (ML13106A331)
5. Exelon Generation Company, LLC, Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (RS-14-070), dated March 31, 2014 (ML14090A241)
6. Exelon Generation Company, LLC Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident – 1.5 Year Response for CEUS Sites (RS-13-205), dated September 12, 2013 (ML13256A070)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a 50.54(f) letter to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of Reference 1 requested each addressee located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation and Screening Report within 1.5 years from the date of Reference 1.

In Reference 2, the Nuclear Energy Institute (NEI) requested NRC agreement to delay submittal of the final CEUS Seismic Hazard Evaluation and Screening Reports so that an update to the Electric Power Research Institute (EPRI) ground motion attenuation model could be completed and used to develop that information. NEI proposed that descriptions of subsurface materials and properties and base case velocity profiles be submitted to the NRC by September 12, 2013, (Reference 6) with the remaining seismic hazard and screening information submitted by March 31, 2014 (Reference 5). NRC agreed with that proposed path forward in Reference 4.

Reference 1 requested that licensees provide interim evaluations and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation. In accordance with the NRC endorsed guidance in Reference 3, the enclosed Expedited Seismic Evaluation Process (ESEP) Report for Oyster Creek Nuclear Generating Station provides the information described in the "ESEP Report" Section 7, of Reference 3 in accordance with the schedule identified in Reference 2.

All equipment evaluated for the ESEP for Oyster Creek Nuclear Generating Station was found to have adequate capacity for the required seismic demand as defined by the Augmented Approach (ESEP) guidance (Reference 3). Therefore, no equipment modifications are required.

This ESEP report transmittal completes regulatory Commitment No. 1 of Reference 5.

No new regulatory commitments result from this transmittal.

If you have any questions regarding this report, please contact Ron Gaston at (630) 657-3359.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 19th day of December 2014.

Respectfully submitted,

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

Enclosure:

Oyster Creek Nuclear Generating Station Expedited Seismic Evaluation Process (ESEP) Report

cc: Director, Office of Nuclear Reactor Regulation
Regional Administrator - NRC Region I
NRC Senior Resident Inspector – Oyster Creek Nuclear Generating Station
NRC Project Manager, NRR – Oyster Creek Nuclear Generating Station
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Manager, Bureau of Nuclear Engineering - New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ

Enclosure

Oyster Creek Nuclear Generating Station

Expedited Seismic Evaluation Process (ESEP) Report

(41 pages)

EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) REPORT

IN RESPONSE TO THE 50.54(f) INFORMATION REQUEST REGARDING FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

for the

Oyster Creek Generating Station
Route 9 South P.O. Box 388
Forked River, New Jersey 08731
Facility Operating License No. DPR-16
NRC Docket No. STN 50-219
Correspondence No.: RS-14-299

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Report Number: 14Q4241-RPT-003, Rev. 5

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Corporate Acceptance:	Jeffrey S. Clark		12/16/14

Document Title: EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) IN RESPONSE TO THE 50.54(f) INFORMATION REQUEST REGARDING FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC for the OYSTER CREEK GENERATING STATION

Document Type:
Criteria ☐ Interface ☐ Report ☒ Specification ☐ Other ☐ Drawing ☐

Project Name: Exelon ESEP for Oyster Creek
Job No.: 14Q4241
Client: Exelon

This document has been prepared in accordance with the S&A Quality Assurance Program Manual, Revision 17 and project requirements:

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Prepared by: Seth Baker	Date: 11/26/2014
Reviewed by: Walter Djordjevic	Date: 11/26/2014
Approved by: Walter Djordjevic	Date: 11/26/2014

Revision Record:

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1	Seth Baker 12/01/2014	Walter Djordjevic 12/01/2014	Walter Djordjevic 12/01/2014	Address minor formatting errors.
2	Seth Baker 12/05/2014	Walter Djordjevic 12/05/2014	Walter Djordjevic 12/05/2014	Incorporate client comments
3	Seth Baker 12/09/2014	Walter Djordjevic 12/09/2014	Walter Djordjevic 12/09/2014	Incorporate additional client comments
4	Seth Baker 12/12/2014	Walter Djordjevic 12/12/2014	Walter Djordjevic 12/12/2014	Incorporate additional client comments
5	Seth Baker 12/16/2014	Walter Djordjevic 12/16/2014	Walter Djordjevic 12/16/2014	Incorporate client comments to Section 7.1

S&A Stevenson & Associates	DOCUMENT APPROVAL SHEET	CONTRACT NO. 14Q4241

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1.0 Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Depending on the comparison between the reevaluated seismic hazard and the current design basis, further risk assessment may be required. Assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report describes the Expedited Seismic Evaluation Process (ESEP) undertaken for Oyster Creek Generating Station. The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter [1] to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is implemented using the methodologies in the NRC endorsed guidance in EPRI 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic [2]. EPRI 3002000704 also contains a scope reduction allowance for low seismic hazard sites with a ground motion response spectrum (GMRS) that only exceeds the safe shutdown earthquake (SSE) at low frequencies. Section 4 of the Oyster Creek Seismic Hazard and Screening Report [4] presents justification for classifying the plant as a low seismic hazard site as well as a discussion of the GMRS to SSE exceedance at low frequencies. The allowed reduction in scope will limit the ESEP to equipment items with potential susceptibility to damage from spectral accelerations at low frequencies.

The objective of this report is to provide summary information describing the ESEP evaluations and results. The level of detail provided in the report is intended to enable NRC to understand the inputs used, the evaluations performed, and the decisions made as a result of the interim evaluations.

2.0 Brief Summary of the FLEX Seismic Implementation Strategies

The Oyster Creek FLEX response strategies to maintain Core Cooling, Containment, Spent Fuel Pool Cooling, and Safety Function Support are summarized below. This summary is derived from the Oyster Creek Overall Integrated Plan (OIP), including all 6 month FLEX updates through August 2014, in Response to the March 12, 2012, NRC Order EA-12 049 [3].

Flex Phase 1, 1.5 hours, strategy relies on installed plant equipment. The Reactor will automatically isolate maintaining RPV inventory. Reactor Core Cooling, and Decay Heat Removal is achieved through the Isolation Condenser System (ICS). The ICS is comprised of two heat exchangers. The ICS is placed into service by opening a single DC powered condensate return valve in each system. The condensate return valves open automatically and are then manually cycled to limit RPV cool down rate. The ICS removes decay heat and deposits it into the environment and not into Containment. The ICS is a closed loop system; RPV inventory is not lost due to ICS operation. Oyster Creek is a hot shutdown design and as long as water is supplied to the ICS shells coping can extend indefinitely. The ICS can provide decay heat removal for 1 hour 40 minutes without makeup water being provided to the condenser shells. Although not credited in the FLEX time line off site redundant fire diesels can provide water to the ICS shells if the fire system was not damaged in the initiating event. The fire protection system is considered a defense in-depth system, use if available, but does not affect the FLEX primary strategy.

Key Reactor Parameters are obtained via DC powered and locally installed instrumentation. A DC load shedding strategy is employed to extend battery life.

No specific Containment Control is required in Phase 1 as both temperature and pressure stay within design limits for the first 72 hours of the event. Key containment parameters are obtained from DC powered instrumentation or from locally installed gauges.

No specific Spent Fuel Pool control is required in Phase 1 as both temperature and level stay within design limits for the first 14.5 hours of the event. Spent Fuel Pool level is obtained from the new Spent Fuel Pool wide range instrumentation installed under order EA-12-051 [20].

No specific Safety Function Support actions are required during phase 1.

Flex Phase 2, 1.5 to 24 hours, strategy relies on installed plant equipment and portable equipment.

Core Cooling is ensured by providing water to the ICS shells within 1.5 hours. Makeup water is supplied using the FLEX pump taking suction from the intake or discharge canal. Reactor Inventory control is managed using a connection to Core Spray System I and uses the same FLEX pump that provides makeup water to ICS shells. The ICS system reduces reactor pressure to the point that the low pressure FLEX pump can inject into the RPV. Reactor Inventory loss and containment energy addition are from reactor recirculation pump seal leakage and unidentified leakage, with the major contributor being recirculation pump seal leakage.

During phase 2, electrical power is restored at 2.5 hours. A portable 500KW 480VAC diesel generator is installed at 480 VAC Unit Substations USS 1A2 or USS 1B2. This re-powered USS will provide power to Battery chargers, ICS MOVs, and Control Rod Drive (CRD) pump if the Condensate Storage Tank (CST) is available. The use of the CRD pump is a FLEX defense in-depth strategy. The CST is not a protected or seismically qualified water source but if available, would provide a clean high pressure injection source to the RPV.

Electrical power is used to isolate reactor recirculation pumps, limiting RPV losses and energy addition to the containment from recirculation pump seal leakage. This restored power is also used to re-power station battery chargers ensuring the continued availability of DC power to provide critical instrumentation and DC valve operation.

Key Reactor Parameters are obtained via DC powered instrumentation or via the 500KW 480 VAC generator to re-power Motor Control Centers (MCCs) required to provide additional instrumentation.

No specific Containment Control is required in Phase 2 as both temperature and pressure stay within design limits for the first 72 hours of the event. Key Containment Parameters are obtained from DC powered instrumentation, local gauges, or instrumentation re-powered from the FLEX generator.

Spent Fuel Pool control is required in Phase 2. At 6 hours a connection from the FLEX pump will be made to the Spent Fuel Pool Cooling (SFPC) systems existing B.5.b connection to provide makeup water to the fuel pool. An alternate SFPC strategy is to provide a water spray directly to the fuel pool on the refueling floor.

Spent Fuel Pool level is obtained from the new Spent Fuel Pool wide range instrumentation installed under order EA-12-051 [20].

Safety Functions Support strategies in phase 2 include the control room, battery room, and refuel floor habitability. The strategies include opening of doors and roof hatches, and the use of portable fans and blowers, to provide ventilation to affected areas.

Flex Phase 3, hour 24 to 72, strategy relies on installed plant equipment and portable equipment.

Phase 1 and 2 strategy will provide sufficient capability that no additional Phase 3 strategies are required. Phase 3 equipment for Oyster Creek includes backup portable pumps and generators. The portable pumps will be capable of providing the necessary flow and pressure as outlined in Phase 2 response for Core Cooling & Decay Heat Removal, RCS Inventory Control and Spent Fuel Pool Cooling. The portable generators will be capable of providing the necessary 480 VAC power requirements as outlined in Phase 2 response for Safety Functions Support.

3.0 Equipment Selection Process and ESEL, Alternate Path Justifications, and Determination of the Reduced ESEL

The selection of equipment for the Expedited Seismic Equipment List (ESEL) followed the guidelines of EPRI 3002000704 [2]. Per the EPRI guidance, a full ESEL [17] was first developed without considering the allowed reduction for low seismic hazard sites having only low frequency exceedance of GMRS to SSE (<2.5 Hz). The full ESEL for Oyster Creek is presented in Attachment A and the reduced ESEL is presented in Attachment B. Section 3.1 and 3.2 of this report detail the selection process for the full ESEL while the selection process for the reduced ESEL is discussed in Section 3.3.

3.1 Equipment Selection Process and ESEL

The selection of equipment on the ESEL was based on installed plant equipment credited in the FLEX strategies during Phase 1, 2 and 3 mitigation of a Beyond Design Basis External Event (BDBEE), as outlined in the Oyster Creek Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049 [3]. The OIP, including 6 month updates through August 2014, provides the Oyster Creek FLEX mitigation strategy and serves as the basis for the equipment selected for the ESEP.

The scope of "installed plant equipment" includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment integrity consistent with the Oyster Creek OIP [3]. FLEX recovery actions are excluded from the ESEP scope per EPRI 3002000704 [2]. The overall list of planned FLEX modifications and the scope for consideration herein is limited to those required to support core cooling, reactor coolant inventory, subcriticality, and containment integrity functions. Portable and pre-staged FLEX equipment (not permanently installed/anchored) are excluded from the ESEL per EPRI 3002000704 [2].

The ESEL component selection followed the EPRI guidance outlined in Section 3.2 of EPRI 3002000704 [2].

1. The scope of components is limited to those required to accomplish the core cooling and containment safety functions identified in Table 3-1 of EPRI 3002000704 [2]. The instrumentation monitoring requirements for core cooling/containment safety functions are limited to those outlined in the EPRI 3002000704 [2] guidance, and are a subset of those outlined in the Oyster Creek OIP [3].

2. The scope of components is limited to installed plant equipment, and FLEX connections necessary to implement the Oyster Creek OIP [3] as described in Section 2.

3. The scope of components assumes the credited FLEX modifications, including connections, are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate").

4. The "Primary" FLEX success path is to be specified. Selection of the "Back-up/Alternate" FLEX success path must be justified if used.

5. Phase 3 coping strategies are included in the ESEP scope, whereas recovery strategies are excluded.

6. Structures, systems, and components excluded per the EPRI 3002000704 [2] guidance are:
   - Structures (e.g. containment, reactor building, control building, auxiliary building, etc.)
   - Piping, cabling, conduit, HVAC, and their supports.
   - Manual valves and rupture disks.
   - Power-operated valves not required to change state as part of the FLEX mitigation strategies.
   - Nuclear steam supply system components (e.g. reactor pressure vessel and internals, reactor coolant pumps and seals, etc.)

7. For cases in which neither train was specified as a primary or back-up strategy, then only one train component (generally 'A' train) is included in the ESEL.

3.1.1 ESEL Development

The ESEL was developed by reviewing the Oyster Creek OIP, including all 6 month FLEX updates through August 2014, [3] to determine the major equipment involved in the FLEX strategies. Further reviews of plant drawings (e.g., Process and Instrumentation Diagrams (P&IDs) and Electrical One Line Diagrams) were performed to identify the boundaries of the flow paths to be used in the FLEX strategies and to identify specific components in the flow paths needed to support implementation of the FLEX strategies. Boundaries were established at an electrical or mechanical isolation device (e.g., isolation amplifier, valve, etc.) in branch circuits / branch lines off the defined electrical or fluid flow path. P&IDs were the primary reference documents used to identify mechanical components and instrumentation. The flow paths used for FLEX strategies were selected and specific components were identified using detailed equipment and instrument drawings, piping isometrics, electrical schematics and one-line diagrams, system descriptions, and design basis documents.

The flow paths credited for the Oyster Creek ESEP are shown in Table 3-1 below.

Table 3-1: Flow Paths Credited for ESEP

Flow Path	FLEX Drawing	P&IDs
Steam from the Reactor Pressure Vessel to the Emergency Condensers and condensate from the Emergency Condensers to the Reactor Recirculation Piping	FLEX Second Six-Month Status Report Attachment 1 [3.3]	GE 148F262 Sh. 1 [19.1] GE 237E798 [19.2]
Make up coolant from the Ultimate Heat Sink to the Emergency Condenser Secondary Side via a FLEX pump and resulting steam vented to Atmosphere	FLEX Second Six-Month Status Report Attachment 1 [3.3]	GE 148F262 Sh. 1 [19.1]
RPV/RCS make up coolant from the Ultimate Heat Sink to Core Spray System via Flex pump connection	FLEX Second Six-Month Status Report Attachment 1 [3.3]	GE 885D781 Sh. 1 [19.3]
Drywell and Torus Hardened Containment Ventilation System, vents structures to atmosphere	None	GU 3E 243-21-1000 Sh. 1 [19.4] BR 2011 Sh. 2 [19.5] SN 13432.19-1 Sh. 1 [19.6]
Coolant from the Ultimate Heat Sink to Containment Spray system via FLEX pump connection to control Containment/Drywell pressure	FLEX Second Six-Month Status Report Attachment 1 [3.3]	GE 148F740 Sh. 1 [19.7]
Isolation of the Reactor Recirculation Pump seals to minimize RPV/RCS leakage	None	GE 237E798 [19.2]
Fuel Oil from the Diesel Generator Fuel Oil Tank to the FLEX Connection Point	FLEX Second Six-Month Status Report Attachment 1 [3.3]	GU 3E-862-21-1000 Sh. 1 [19.8]

3.1.2 Power Operated Valves

Page 3-3 of EPRI 3002000704 [2] notes that power operated valves not required to change state are excluded from the ESEL. Page 3-2 also notes that "functional failure modes of electrical and mechanical portions of the installed Phase 1 equipment should be considered (e.g. RCIC/AFW trips)". To address this concern, the following guidance is applied in the Oyster Creek ESEL for functional failure modes associated with power operated valves:

- Power operated valves that must remain energized during the Extended Loss of all AC Power (ELAP) events in order to maintain a credited FLEX flow path or pressure boundary (such as DC powered solenoid-operated valves), were included on the ESEL.
- Power operated valves not required to change state as part of the FLEX mitigation strategies were not included on the ESEL. The seismic event also causes the ELAP event; therefore, the valves are incapable of spurious operation as they would be de-energized.
- Power operated valves not required to change state as part of the FLEX mitigation strategies during Phase 1, and are re-energized and operated during subsequent Phase 2 and 3 strategies, were not evaluated for spurious valve operation as the seismic event that caused the ELAP has passed before the valves are re-powered.

3.1.3 Pull Boxes

Pull boxes were deemed unnecessary to add to the ESEL as these components provide completely passive locations for pulling or installing cables. No breaks or connections in the cabling are included in pull boxes. Pull boxes were considered part of conduit and cabling, which are excluded in accordance with EPRI 3002000704 [2].

3.1.4 Termination Cabinets

Termination cabinets, including cabinets necessary for FLEX Phase 2 and Phase 3 connections, provide consolidated locations for permanently connecting multiple cables. The termination cabinets and the internal connections provide a completely passive function; however, the cabinets are included in the ESEL to ensure industry knowledge on panel/anchorage failure vulnerabilities is addressed and the connections are excluded from the ESEL.

3.1.5 Critical Instrumentation Indicators

Critical indicators and recorders are typically physically located on panels/cabinets and are included as separate components; however, seismic evaluation of the instrument indication may be included in the panel/cabinet seismic evaluation (rule-of-the-box).

3.1.6 Phase 2 and Phase 3 Piping Connections

Item 2 in Section 3.1 above notes that the scope of equipment in the ESEL includes "... FLEX connections necessary to implement the Oyster Creek OIP [3] as described in Section 2."

Item 3 in Section 3.1 notes that "The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate")."

Item 6 in Section 3 goes on to explain that "Piping, cabling, conduit, HVAC, and their supports" are excluded from the ESEL scope in accordance with EPRI 3002000704 [2].

Therefore, piping and pipe supports associated with FLEX Phase 2 and Phase 3 connections are excluded from the scope of the ESEP evaluation. However, any active valves in the FLEX Phase 2 and Phase 3 connection flow path are included in the ESEL.

3.2 Justification for use of Equipment that is not the Primary Means for FLEX Implementation

All equipment used for FLEX implementation on the Oyster Creek ESEL are primary path.

3.3 Determination of the Reduced ESEL

EPRI 3002000704 [2] contains an ESEL reduction allowance for plants qualifying as low seismic hazard sites under Section 3.2.1.1 of EPRI 1025287 [14]. This provision allows qualifying plants to limit the ESEL to equipment that are potentially susceptible to damage from spectral accelerations at low frequencies. Section 4 of the Oyster Creek Seismic Hazard and Screening Report [4] presents justification for classifying the plant as a low seismic hazard site. An excerpt from the Oyster Creek Seismic Hazard and Screening Report [4] is shown below. Refer to Section 4.0 for plots of the Oyster Creek SSE and GMRS.

> *In the frequency range of 1 to 10Hz, the Oyster Creek Nuclear Generating Station (OCNGS) SSE spectral acceleration exceeds that of the GMRS except for frequencies below approximately 1.9 Hz [4]. According to the Screening, Prioritization and Implementation Details (SPID), Section 3.2.1.1 [14], the OCNGS SSE exceedances of the GMRS in the frequency range of 1 to 10Hz are classified as low-frequency exceedances. Further, the GMRS spectral acceleration does not exceed the low hazard threshold of 0.4g peak spectral acceleration. For most Structures, Systems, and Components (SSCs), exceedances below 2.5 Hz are non-consequential as the fundamental frequency of these SSCs exceeds 2.5 Hz. Because of this and the low likelihood of any seismically designed SSC being damaged by ground motion with a peak spectral acceleration less than the low hazard threshold, the expected seismic risk at OCNGS is low [14]. As a result, the SPID, Section 3.2.1.1 [14] limits the seismic risk assessment to evaluation of safety-significant SSCs that are potentially susceptible to ground motions at frequencies less than 1.9 Hz for OCNGS.*
>
> *Examples of SSCs and failure modes potentially susceptible to damage from spectral accelerations at low frequencies are provided in the SPID, Section 3.2.1.1 [14] and reproduced below. Based upon further review of equipment natural frequencies, an additional component type was identified as potentially susceptible to low frequency*

acceleration: equipment mounted on vibration isolators. The SSC and failure mode types, along with examples of specific potentially safety-significant OCNGS SSCs, are listed below.

- *Liquid sloshing in atmospheric pressure storage tanks*
  - *Diesel generator fuel oil storage tank, T-39-2*
  - *Condensate storage tank, T-11-1*
- *Very flexible distribution systems with frequencies less than 1.9 Hz*
  - *Cable tray raceways*
  - *Conduit raceways*
  - *Flexible piping systems*
- *Sliding and rocking of unanchored components*
  - *Emergency diesel generators, M-39-001 and M-39-002*
  - *Fire water pump house (controlling failure mode is sliding)*
- *Fuel assemblies inside the reactor vessel*
- *Soil liquefaction*
  - *Emergency diesel generator building*
  - *Turbine building*
  - *Fire water buried piping*
- *Equipment mounted on vibration isolators*
  - *Batt & M-G room exhaust and supply fans, EF-1-20 and SF-1-20*
  - *Switchgear room "A" main exhaust and supply fans, FN-56-4 and FN-56-7*

The above Structures, Systems, and Components (SSCs) were compared against the full ESEL presented in Attachment A. The only overlapping item is the diesel generator fuel oil storage tank, tag number T-39-2. Per Section 2.2.1.1 of EPRI 3002000704 [2] the ESEL is therefore reduced to only include tank T-39-2, as presented in Attachment B.

4.0 Ground Motion Response Spectrum (GMRS)

4.1 Plot of GMRS Submitted by the Licensee

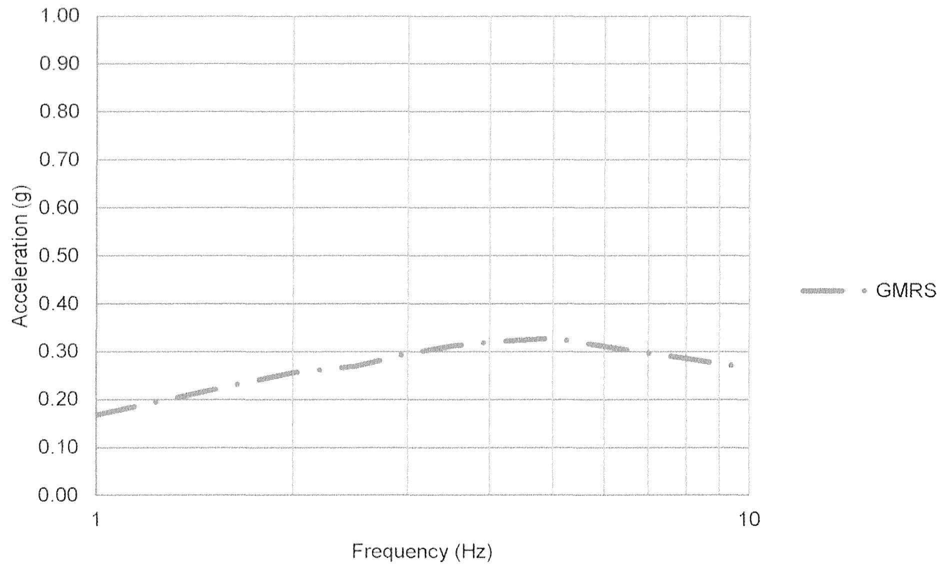
In accordance with Section 2.4.2 of the Screening, Prioritization and Implementation Details (SPID) [14], the licensing design basis definition of the SSE control point for Oyster Creek is used for comparison to the GMRS. The Oyster Creek March 31, 2014 Submittal [4] states that the site SSE, anchored to a peak ground acceleration (PGA) of 0.184g, is defined at elevation 23 feet.

The GMRS, taken from the Oyster Creek March 31, 2014 Submittal report [4] is shown in Table 4-1 and Figure 4-1.

Table 4-1: Oyster Creek GMRS (5% Damping)

Freq. (Hz)	GMRS (unscaled, g)
1	0.168
1.25	0.196
1.5	0.220
2	0.256
2.5	0.270
3	0.296
3.5	0.312
4	0.320
5	0.328
6	0.311
7	0.297
8	0.286
9	0.275
10	0.266

Figure 4-1: Oyster Creek GMRS Plot (5% Damping)



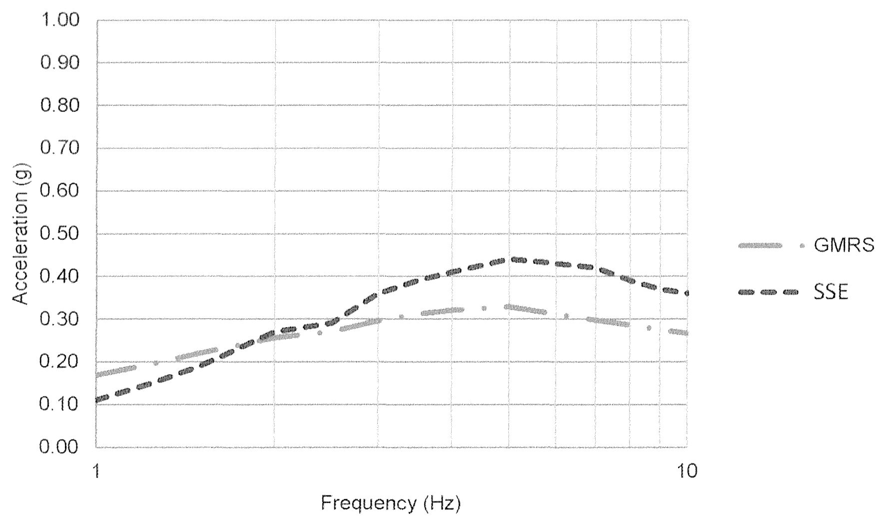
4.2 Comparison to SSE

As identified in the March submittal report, the GMRS only exceeds the SSE below 1.9 Hz within the 1-10 Hz range. A comparison of the GMRS to the SSE between 1-10Hz is shown in Table 4-2 and Figure 4-2. Per EPRI 3002000704 [2], low-frequency GMRS exceedances (below 2.5 Hz) at low seismic hazard sites do not require a plant to perform a full ESEP.

Table 4-2: Oyster Creek GMRS vs. SSE (5% Damping)

Freq. (Hz)	GMRS (unscaled, g)	Horizontal SSE (g)
1	0.168	0.110
1.25	0.196	0.150
1.5	0.220	0.190
2	0.256	0.270
2.5	0.270	0.290
3	0.296	0.360
3.5	0.312	0.390
4	0.320	0.410
5	0.328	0.440
6	0.311	0.430
7	0.297	0.420
8	0.286	0.390
9	0.275	0.370
10	0.266	0.360

Figure 4-2: Oyster Creek GMRS vs. SSE Plot (5% Damping)



5.0 Review Level Ground Motion (RLGM)

5.1 Description of RLGM selected

The RLGM for Oyster Creek was determined in accordance with Section 4 of EPRI 30020000704 [2] by linearly scaling the SSE by the maximum GMRS/SSE ratio between the 1 and 10 Hz range. This calculation is shown in Table 5-1.

Table 5-1: Ratio Between GMRS AND SSE (5% Damping)

Freq. (Hz)	GMRS (unscaled, g)	Horizontal SSE (g)	GMRS/SSE
1	0.168	0.110	1.53
1.25	0.196	0.150	1.31
1.5	0.220	0.190	1.16
2	0.256	0.270	0.95
2.5	0.270	0.290	0.93
3	0.296	0.360	0.82
3.5	0.312	0.390	0.80
4	0.320	0.410	0.78
5	0.328	0.440	0.75
6	0.311	0.430	0.72
7	0.297	0.420	0.71
8	0.286	0.390	0.73
9	0.275	0.370	0.74
10	0.266	0.360	0.74

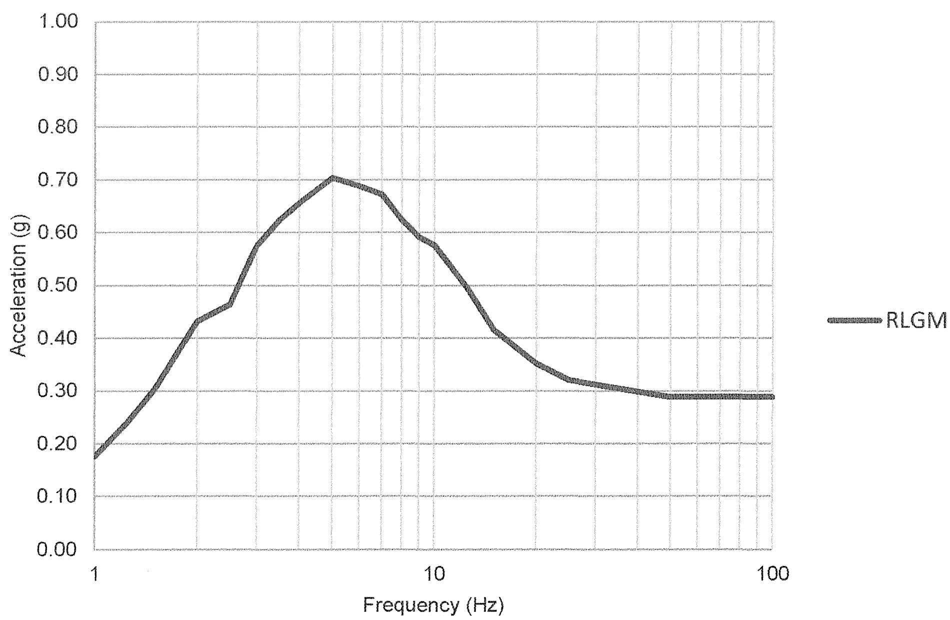
As shown above, the maximum GMRS/SSE ratio for Oyster Creek occurs at 1.0 Hz and equals 1.53, which is conservatively rounded up to 1.60.

The resulting 5% damped RLGM, based on scaling the horizontal SSE by the scale factor of 1.60, is shown below in Table 5-2 and Figure 5-1 below. Note that the RLGM peak ground acceleration (PGA) is 0.29g. Seismic capacities for equipment will be compared against the PGA of the RLGM.

Table 5-2: RLGM (5% Damping)

Freq. (Hz)	RLGM (g)
1	0.18
1.25	0.24
1.5	0.30
2	0.43
2.5	0.46
3	0.58
3.5	0.62
4	0.66
5	0.70
6	0.69
7	0.67
8	0.62
9	0.59
10	0.58
12.5	0.50
15	0.42
20	0.35
25	0.32
50	0.29
100	0.29

Figure 5-1: Plot of RLGM (5% Damping)



5.2 Method to Estimate ISRS

The method used to derive the ESEP in-structure response spectra (ISRS) was to uniformly scale existing SSE-based ISRS from 50124-R-001 [16] by the maximum scale factor of 1.60 from Table 5-1. Scaled ISRS are calculated for all locations where ESEL items are located at Oyster Creek. These scaled ISRS are documented within calculation 14Q4241-CAL-001 [10].

6.0 Seismic Margin Evaluation Approach

It is necessary to demonstrate that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM. The seismic capacity is characterized as the highest PGA for which there is a high confidence of a low probability of failure (HCLPF). The PGA is associated with a particular spectral shape, in this case the 5%-damped RLGM spectral shape. The calculated HCLPF capacity must be equal to or greater than the RLGM PGA (0.290g from Table 5-2). The criteria for seismic capacity determination are given in Section 5 of EPRI 3002000704 [2].

There are two basic approaches for developing HCLPF capacities:

1. Deterministic approach using the conservative deterministic failure margin (CDFM) methodology of EPRI NP-6041 [7].
2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959 [8].

For Oyster Creek, the deterministic approach using the CDFM methodology of EPRI NP-6041 [7] was used to determine HCLPF capacities.

6.1 Summary of Methodologies Used

Oyster Creek performed a probabilistic risk assessment (PRA) that was concluded in 2001. The PRA is documented in the Oyster Creek IPEEE report [9] and consisted of walkdowns and HCLPF calculations. The walkdowns were conducted by engineers trained in EPRI NP-6041 and PRA. Walkdown results were documented on Screening Evaluation Work Sheets (SEWS) from EPRI NP-6041 [7] in concert with the Unresolved Safety Issue (USI) A-46 evaluation of Oyster Creek.

The screening walkdowns used Table 2-4 of EPRI NP-6041 [7]. The walkdowns were conducted by engineers who as a minimum attended the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation Training Course. The walkdowns were documented on Screening Evaluation Work Sheets (contained within report 14Q4241-RPT-005 [10]) from EPRI NP-6041 [7]. Anchorage capacity calculations used the CDFM criteria from EPRI NP-6041 [7]. The input seismic demand was RLGM shown in Table 5-2 and Figure 5-1.

6.2 HCLPF Screening Process

The spectral peak RLGM for Oyster Creek reaches approximately 0.70g at 5 Hz (Table 5-2:). The screening tables in EPRI NP-6041 [7] are based on ground peak spectral accelerations of 0.8g and 1.2g. These both exceed the RLGM peak spectral acceleration. The Oyster Creek reduced ESEL components were screened against the 0.8g column of Table 2-4 of NP-6041.

The Oyster Creek reduced ESEL (Attachment B) contains one item: the diesel generator fuel oil storage tank, tag number T-39-2. In accordance with Table 2-4 of EPRI NP-6041 [7], all atmospheric storage tanks require HLCPF evaluation. The HCLPF evaluation for tank T-39-2 is performed within calculation 14Q4241-CAL-002 [10], and results are summarized in Attachment C of this report. HCLPF capacities are compared against the RGLM peak ground acceleration identified in Section 5.1.

6.3 Seismic Walkdown Approach

6.3.1 Walkdown Approach

Walkdowns for Oyster Creek were performed in accordance with the criteria provided in Section 5 of EPRI 3002000704 [2], which refers to EPRI NP-6041 [7] for the Seismic Margin Assessment process. Pages 2-26 through 2-30 of EPRI NP-6041 [7] describe the seismic walkdown criteria, including the following key criteria.

> *"The SRT [Seismic Review Team] should "walk by" 100% of all components which are reasonably accessible and in non-radioactive or low radioactive environments. Seismic capability assessment of components which are inaccessible, in high-radioactive environments, or possibly within contaminated containment, will have to rely more on alternate means such as photographic inspection, more reliance on seismic reanalysis, and possibly, smaller inspection teams and more hurried inspections. A 100% "walk by" does not mean complete inspection of each component, nor does it mean requiring an electrician or other technician to de-energize and open cabinets or panels for detailed inspection of all components. This walkdown is not intended to be a QA or QC review or a review of the adequacy of the component at the SSE level.*
>
> *If the SRT has a reasonable basis for assuming that the group of components are similar and are similarly anchored, then it is only necessary to inspect one component out of this group. The "similarity-basis" should be developed before the walkdown during the seismic capability preparatory work (Step 3) by reference to drawings, calculations or specifications. The one component of each type which is selected should be thoroughly inspected which probably does mean de-energizing and opening cabinets or panels for this very limited sample. Generally, a spare representative component can be found so as to enable the inspection to be performed while the plant is in operation. At least for the one component of each type which is selected, anchorage should be thoroughly inspected.*
>
> *The walkdown procedure should be performed in an ad hoc manner. For each class of components the SRT should look closely at the first items and compare the field configurations with the construction drawings and/or specifications. If a one-to-one correspondence is found, then subsequent items do not have to be inspected in as great a detail. Ultimately the walkdown becomes a "walk by" of the component class as the SRT becomes confident that the construction pattern is typical. This procedure for inspection should be repeated for each component class; although, during the actual walkdown the SRT may be inspecting several classes of components in parallel. Is serious exceptions to the drawings or questionable construction practices are found then the system or component class must be inspected in closer detail until the systematic deficiency is defined.*

> *The 100% "walk by" is to look for outliers, lack of similarity, anchorage which is different from that shown on drawings or prescribed in criteria for that component, potential SI [Seismic Interaction][1] problems, situations that are at odds with the team members' past experience, and any other areas of serious seismic concern. If any such concerns surface, then the limited sample size of one component of each type for thorough inspection will have to be increased. The increase in sample size which should be inspected will depend upon the number of outliers and different anchorages, etc., which are observed. It is up to the SRT to ultimately select the sample size since they are the ones who are responsible for the seismic adequacy of all elements which they screen from the margin review. Appendix D gives guidance for sampling selection.*

As shown in Attachment B, the only item on the reduced ESEL is the diesel generator fuel oil storage tank, tag number T-39-2. However, the SRT deemed it prudent to walk down an expanded set of equipment that could potentially be susceptible to damage from low frequency spectral accelerations, namely the motor control centers, battery racks, and isolation condensers listed in Attachment A. Upon visual inspection, the SRT judged these items to have a natural frequency well above 1.9 Hz (the GMRS-SSE intersection point) and confirmed that they could be excluded from the reduced ESEL.

The diesel generator fuel oil storage tank is located within a confined space and the SRT was not permitted to access the area during the time of the walkdown. The tank was previously walked down during NTTF 2.3 [15] and USI A-46 [18] and it was determined that enough preexisting information was available to preclude the need to enter the confined space around the tank. Furthermore, EPRI 3002000704 [2] limits the ESEP seismic interaction reviews to nearby block walls and piping attached to tanks[1]. Given that no block walls exist within the tank enclosure and previous walkdown information shows that piping exhibits adequate flexibility, a future walkdown to check tank seismic interactions is not necessary. Previous walkdown information that was relied upon is documented in Section 6.3.2.

6.3.2 Application of Previous Walkdown Information

As discussed in Section 6.3.1, the confined space around the diesel generator fuel oil storage tank (T-39-2) prevented access during the time of the walkdowns. Previous walkdown information from NTTF 2.3 [15], along with existing calculations and SEWS from the USI A-46 evaluation [18], were determined to provide a sufficient amount of information for the purposes of ESEP.

6.3.3 Significant Walkdown Findings

Consistent with the guidance from NP-6041 [7], no significant outliers or anchorage concerns were identified during the Oyster Creek ESEP walkdowns.

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[1] EPRI 3002000704 [2] page 5-4 limits the ESEP seismic interaction reviews to "nearby block walls" and "piping attached to tanks" which are reviewed "to address the possibility of failures due to differential displacements." Other potential seismic interaction evaluations are "deferred to the full seismic risk evaluations performed in accordance with EPRI 1025287 [14]."

6.4 HCLPF Calculation Process

ESEL items were evaluated using the criteria in EPRI NP-6041 [7]. Those evaluations included the following steps:

- Performing seismic capability walkdowns for equipment to evaluate the equipment installed plant conditions
- Performing screening evaluations using the screening tables in EPRI NP-6041 as described in Section 6.2
- Performing HCLPF calculations considering various failure modes that include both structural (e.g. anchorage, load path etc.) and functional failure modes.

HCLPF calculations were performed using the CDFM methodology and are documented in calculation 14Q4241-CAL-002 [10], with results summarized in Attachment C of this report. HCLPF capacities are compared against the RGLM peak ground acceleration identified in Section 5.1.

The CDFM analysis criteria established in Section 6 of EPRI NP-6041 [7] are used when detailed analysis is required. The relevant CDFM criteria from EPRI NP-6041 [7] are summarized in Table 6-1.

Table 6-1: HCLPF Calculation Summary

Load combination:	Normal + Ec
Ground response spectrum:	Conservatively specified (84% non-exceedance probability)
Damping:	Conservative estimate of median damping.
Structural model:	Best estimate (median) + uncertainty variation in frequency.
Soil-structure interaction	Best estimate (median) + parameter variation
Material strength:	Code specified minimum strength or 95% exceedance of actual strength if test data is available.
Static capacity equations:	Code ultimate strength (ACI), maximum strength (AISC), Service Level D (ASME) or functional limits. If test data is available to demonstrate excessive conservatism of code equations then use 84% exceedance of test data for capacity equations.
Inelastic energy absorption:	For non-brittle failure modes and linear analysis, use 80% of computed seismic stress in capacity evaluation to account for ductility benefits or perform nonlinear analysis and use 95% exceedance ductility levels.
In-structure (floor) spectra generation:	Use frequency shifting rather than peak broadening to account for uncertainty and use median damping.

The HCLPF capacity is equal to the PGA at which the strength limit is reached. The HCLPF earthquake load is calculated as follows:

U = Normal + Ec

Where:

- U = Ultimate strength per Section 6 of EPRI NP-6041 [7]
- Ec = HCLPF earthquake load
- Normal = Normal operating loads (dead and live load expected to be present, etc.)

For this calculation, the HCLPF earthquake load is related to a fixed reference earthquake:

Ec = SFc*Eref

Where:

- Eref = reference earthquake from the relevant in-structure response spectrum (ISRS)
- SFc = component-specific scale factor that satisfies U = Normal +Ec

The HCLPF will be defined as the PGA produced by Ec. The Oyster Creek RLGM PGA is 0.290g, therefore:

HCLPF = 0.290g*SFc

6.5 Functional evaluation of relays

Relays are not considered vulnerable to low frequency spectral accelerations and therefore do not need to be included in the reduced ESEL per section 2.2.1.1 of EPRI 3002000704 [2].

6.6 Tabulated ESEL HCLPF Values (Including Key Failure Modes)

Tabulated ESEL HCLPF values including key failure modes for low frequency ESEL items are included in Attachment C. Anchorage failure controls the diesel generator fuel oil storage tank HCLPF; therefore, the anchorage HCLPF value is listed in the table and the failure mode is set to "Anchorage".

7.0 Inaccessible Items

7.1 Identification of ESEL Items Inaccessible for Walkdowns

As discussed in Section 6.3.2, the confined space around the diesel generator fuel oil storage tank (T-39-2) prevented access during the time of the walkdowns. Previous walkdown information from NTTF 2.3 [15] and USI A-46 [18] was determined by the Seismic Review Team (SRT) to provide sufficient information for the purposes of ESEP. Detailed analysis performed in 14Q4241-CAL-002 [10] found the tank (T-39-2) to be acceptable. A future walkdown of tank T-39-2 is not required.

7.2 Planned Walkdown / Evaluation Schedule / Close Out

No additional walkdowns are required.

8.0 ESEP Conclusions and Results

8.1 Supporting Information

Oyster Creek Generating Station has performed the ESEP as an interim action in response to the NRC's 50.54(f) letter [1]. It was performed using the methodologies in the NRC endorsed guidance in EPRI 3002000704 [2].

The ESEP provides an important demonstration of seismic margin and expedites plant safety enhancements through evaluations and potential near-term modifications of plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is part of the overall Oyster Creek response to the NRC's 50.54(f) letter [1]. On March 12, 2014, NEI submitted to the NRC results of a study [12] of seismic core damage risk estimates based on updated seismic hazard information as it applies to operating nuclear reactors in the Central and Eastern United States (CEUS). The study concluded that "site-specific seismic hazards show that there has not been an overall increase in seismic risk for the fleet of U.S. plants" based on the re-evaluated seismic hazards. As such, the "current seismic design of operating reactors continues to provide a safety margin to withstand potential earthquakes exceeding the seismic design basis."

The NRC's May 9, 2014 NTTF 2.1 Screening and Prioritization letter [13] concluded that the "fleetwide seismic risk estimates are consistent with the approach and results used in the GI-199 safety/risk assessment." The letter also stated that "As a result, the staff has confirmed that the conclusions reached in GI-199 safety/risk assessment remain valid and that the plants can continue to operate while additional evaluations are conducted."

An assessment of the change in seismic risk for Oyster Creek was included in the fleet risk evaluation submitted in the March 12, 2014 NEI letter [12] therefore, the conclusions in the NRC's May 9 letter [13] also apply to Oyster Creek.

In addition, the March 12, 2014 NEI letter [12] provided an attached "Perspectives on the Seismic Capacity of Operating Plants," which (1) assessed a number of qualitative reasons why the design of Structures, Systems, and Components (SSCs) inherently contain margin beyond their design level, (2) discussed industrial seismic experience databases of performance of industry facility components similar to nuclear SSCs and (3) discussed earthquake experience at operating plants.

The fleet of currently operating nuclear power plants was designed using conservative practices, such that the plants have significant margin to withstand large ground motions safely. This has been borne out for those plants that have actually experienced significant earthquakes. The seismic design process has inherent (and intentional) conservatisms which result in significant seismic margins within SSCs. These conservatisms are reflected in several key aspects of the seismic design process, including:

- Safety factors applied in design calculations
- Damping values used in dynamic analysis of SSCs
- Bounding synthetic time histories for in-structure response spectra calculations
- Broadening criteria for in-structure response spectra
- Response spectra enveloping criteria typically used in SSC analysis and testing applications

- Response spectra based frequency domain analysis rather than explicit time history based time domain analysis
- Bounding requirements in codes and standards
- Use of minimum strength requirements of structural components (concrete and steel)
- Bounding testing requirements, and
- Ductile behavior of the primary materials (that is, not crediting the additional capacity of materials such as steel and reinforced concrete beyond the essentially elastic range, etc.).

These design practices combine to result in margins such that the SSCs will continue to fulfill their functions at ground motions well above the SSE.

8.2 Summary of ESEP Identified and Planned Modifications

The results of the Oyster Creek ESEP performed as an interim action in response to the NRC's 50.54(f) letter [1] using the methodologies in the NRC endorsed guidance in EPRI 3002000704 [2] show that evaluated equipment are adequate in resisting the seismic loads expected to result from the site RLGM. Therefore, no plant modifications are required as a result of the Oyster Creek ESEP.

8.3 Modification Implementation Schedule

No modification implementation schedule is required because no modifications are required.

8.4 Summary of Regulatory Commitments

No regulatory commitments are required.

9.0 References

1 NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012.

2 Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 – Seismic. EPRI, Palo Alto, CA: May 2013. 3002000704.

3 Order Number EA-12-049 responses:

3.1 NRC Letter RS-13-023 from Oyster Creek (ML13060A126), "Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)", February 28, 2013

3.2 NRC Letter RS-13-125 from Oyster Creek (ML13240A263), "First Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)", August 28, 2013

3.3 NRC Letter RS-14-013 from Oyster Creek (ML14059A220), "Second Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)", February 28, 2014

3.4 NRC Letter RS-14-211 from Oyster Creek (ML14241A253), "Third Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)", August 28, 2014

4 Oyster Creek Seismic Hazard and GMRS Submittal, Correspondence No. RS-14-070, dated March 31, 2014.

5 Nuclear Regulatory Commission, NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991

6 Nuclear Regulatory Commission, Generic Letter No. 88-20 Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f), June 1991

7 A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Rev. 1, August 1991, Electric Power Research Institute, Palo Alto, CA. EPRI NP 6041

8 Methodology for Developing Seismic Fragilities, August 1991, EPRI, Palo Alto, CA. 1994, TR-103959

9 Staff Evaluation Report of Individual Plant Examination of External Events (IPEEE) submittal for the Oyster Creek Nuclear Generating Station, dated February 8, 2001

10 Oyster Creek ESEP Calculations:

10.1 S&A Calculation 14Q4241-CAL-001 Rev. 1, Generation of In-Structure Response Spectra for use in ESEP Evaluations

10.2 S&A Calculation 14Q4241-CAL-002 Rev. 2, HCLPF Seismic Capacity of Diesel Oil Storage Tank

10.3 S&A Report 14Q4241-RPT-005, Rev. 5, Oyster Creek ESEP SEWS

11 Nuclear Regulatory Commission, NUREG/CR-0098, Development of Criteria for Seismic Review of Selected Nuclear Power Plants, published May 1978

12 Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Seismic Core Damage Risk Estimates Using the Updated Seismic Hazards for the Operating Nuclear Plants in the Central and Eastern United States", March 12, 2014

13 NRC (E Leeds) Letter to All Power Reactor Licensees et al., "Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident," May 9, 2014.

14 Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. EPRI, Palo Alto, CA: February 2013. 1025287.

15 Oyster Creek NTTF 2.3 Seismic Walkdown Submittal, Correspondence No. RS-12-177, dated November 19, 2012

16 Report No. 50124-R-001, Rev. 0, In-Structure Response Spectra for the Oyster Creek Nuclear Generating Station Compilation of Response Spectra for Use in USI A-46 Program

17 S&A Report 14Q4241-RPT-004 Rev. 1, Validation of Expedited Seismic Equipment List

18 A-46 Seismic Qualification SQ-OC-T-39-002 Rev. 1, Diesel Oil Storage Tank T-39-002

19 Oyster Creek P&IDs:

19.1 GE 148F262 Sheet 1, Rev. 55, *Emergency Condenser Flow Diagram*

19.2 GE 237E798, Rev. 36, *Recirculation System Flow Diagram*

19.3 GE 885D781 Sheet 1, Rev. 73, *Core Spray System Flow Diagram*

19.4 GU 3E-243-21-1000 Sheet 1, Rev. 29, *Drywell and Torus Vacuum Relief System Flow Diagram*

19.5 BR 2011 Sheet 2, Rev. 62, *Reactor Building Ventilation Flow Diagram*

19.6 SN 13432.19-1 Sheet 1, Rev. 33, *Nitrogen Supply System Flow Diagram*

19.7 GE 148F740 Sheet 1, Rev. 44, *Containment Spray System Flow Diagram*

19.8 GU 3E-862-21-1000 Sheet 1, Rev. 24, *Emergency Diesel Generator Diesel Fuel Oil Storage & Transfer System Flow Diagram*

20 NRC Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation"

Attachment A: Oyster Creek ESEL

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
1	USS 1A2	480VAC Vital Reactor Bldg Bus A	In Service	In Service	
2	MCC 1A21	Power to Recirculation Loop Isolation Valves	In Service	In Service	
3	MCC 1A21A	Power to Recirculation Loop Isolation Valves	In Service	In Service	
4	VMCC 1A2	Vital Motor Control Center 1A2	In Service	In Service	
5	BTCHG C1	C Station Battery Solid State Static Charger C1	In Service	In Service	
6	Battery Bank C	Vital Bank C Station Battery	In Service	In Service	
7	DC-C 125V	125VDC Distribution Center C	In Service	In Service	
8	DC-F	125VDC Power Panel DC-F	In Service	In Service	Isolation Condenser, Core Spray and EMRV control/logic power
9	DC-2 125VDC	125VDC Motor Control CTR for Reactor Building	In Service	In Service	
10	CD-14-1B	B Isolation Condenser (NE01B)	Standby	In Service as required	Passive component
11	LT-IG0006B	B Isolation Condenser Shell Level XMITR	In Service	In Service	The indicator for this transmitter is located in panel 1F/2F
12	LI-211-1215	B Isolation Condenser Local Shell Level Indication	In Service	In Service	Mechanical instrument
13	V-14-35	B Isolation Condenser Condensate Return Valve	Closed	Open/Closed	
14	USS 1B2	480VAC Vital Reactor Bldg Bus B	In Service	In Service	
15	VMCC 1B2	Vital Motor Control Center 1B2	In Service	In Service	
16	VMCC 1AB2	Vital Motor Control Center 1AB2 (Recirculation Pump Isolation Valve Power)	In Service	In Service	ATS 1AB2 is contained in VMCC 1AB2
17	MCC 1B21A	Power to Recirculation Loop Isolation Valves	In Service	In Service	
18	MCC 1B21	Power to Static Charger and Recirculation Pump Isolation Valves	In Service	In Service	
19	STATIC CHGR	A/B Station Batteries Solid State Static Charger	In Service	In Service	
20	Battery Bank B	Vital Bank B Station Battery (Lead Acid)	In Service	In Service	
21	DC-B 125V	125VDC Distribution Panel B	In Service	In Service	
22	DC-D	125VDC Power Panel D	In Service	In Service	Isolation Condenser, Core Spray and EMRV control/logic power

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
23	DC-1 125VDC	125VDC Isolation Valves Motor Control Center	In Service	In Service	ATS DC-1 is contained in MCC DC-1
24	CD-14-1A	A Isolation Condenser (NE01A)	Standby	In Service as required	Passive component
25	LT-IG0006A	A Isolation Condenser Shell Level XMITR	In Service	In Service	The indicator for this transmitter is located in panel 1F/2F
26	LI-211-1214	A Isolation Condenser Local Shell Level Indication	In Service	In Service	Mechanical Indicator
27	V-14-34	A Isolation Condenser Condensate Return Valve	Closed	Open/Closed	
28	V-20-15	Core Spray to Reactor Parallel Valve System 1	Closed	Open	AC powered valve which will be manually operated during ELAP
29	RK-3	Instrument Rack 03	In Service	In Service	Contains separately powered PT-IP0007 instrument transmitter
30	PT-IP0007	Containment Pressure Transmitter	In Service	In Service Phase 2	
31	1F/2F	MCR Control Reactor & Drywell Cooling Panel	In Service	In Service	
32	5F/6F	Main Control Room Panel 5F/6F	In Service	In Service	Contains separately powered instrument indicators
33	RSP	Remote Shutdown Panel	In Service	In Service	Contains power supplies for, and elements of, credited instruments
34	16R	Containment H2/02 Panel	In Service	In Service	Monitors containment parameters
35	18R	Main Control Room Panel 18R Reactor Protection	In Service	In Service	Contains instruments from the IOP
36	11F	MCR Panel 11F	In Service	In Service	Routes power to panel 12XR via internal fuse 6F7
37	V-23-13	Drywell N2 Purge Valve/Containment Isolation Valve for Hardened Vent	In Service	In Service	

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
38	V-23-14	Drywell N2 Purge Valve/Containment Isolation Valve for Hardened Vent	In Service	In Service	
39	V-23-15	Torus N2 Purge Valve/Containment Isolation Valve for Hardened Vent	In Service	In Service	
40	V-23-16	Torus N2 Purge Valve/Containment Isolation Valve for Hardened Vent	In Service	In Service	
41	DPT-622-1009	Reactor Fuel Zone Level Wide Range I Transmitter (Channel C)	Standby	In Service	The indicator for this transmitter is located in panel 5F/6F
42	PT-622-1018	Reactor Wide Range Pressure Transmitter (Channel C)	Standby	In Service	The indicator for this transmitter is located in panel 5F/6F
43	T-39-2	Diesel Generator Fuel Oil Storage Tank	Standby	Standby	Passive Component
44	V-37-09	Reactor Recirculation Pump NG01-A-Suction Isolation Valve	Open	Closed	
45	V-37-10	Reactor Recirculation Pump NG01-A Discharge Isolation Valve	Open	Closed	
46	V-37-11	Reactor Recirculation Loop 'A' Bypass Valve NG08-A	Open	Closed	
47	V-37-20	Reactor Recirculation Pump NG01-B Suction Isolation Valve	Open	Closed	
48	V-37-21	Reactor Recirculation Pump NG01-B Discharge Isolation Valve	Open	Closed	
49	V-37-22	Reactor Recirculation Loop 'B' Bypass Valve NG08-B	Open	Closed	
50	V-37-31	Reactor Recirculation Pump NG01-C Suction Isolation Valve	Open	Closed	
51	V-37-32	Reactor Recirculation Pump NG01-C Discharge Isolation Valve	Open	Closed	
52	V-37-33	Reactor Recirculation Loop 'C' Bypass Valve NG08-C	Open	Closed	
53	V-37-42	Reactor Recirculation Pump NG01-D Suction Isolation Valve	Open	Closed	
54	V-37-43	Reactor Recirculation Pump NG01-D Discharge Isolation Valve	Open	Closed	
55	V-37-44	Reactor Recirculation Loop 'D' Bypass Valve NG08-D	Open	Closed	
56	V-37-53	Reactor Recirculation Pump NG01-E Suction Isolation Valve	Open	Closed	
57	V-37-54	Reactor Recirculation Pump NG01-E Discharge Isolation Valve	Open	Closed	
58	V-37-55	Reactor Recirculation Loop 'E' Bypass Valve NG08-E	Open	Closed	

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
59	LSP-1AB2	Local Shutdown Panel	Standby	Standby	Contains elements of control for valve V-37-54
60	3F	Panel	In Service	In Service	Contains control switches for recirculation pump valves. Valve control power provided from MCC via internal control transformer
61	IP-4	120V AC Vital Power Distribution Panel	In service	In service	Provides power for credited instruments
62	IT-4	Automatic Transfer Switch	In Service	In Service	Provides power for 120V AC vital power distribution panel IP-4
63	IT-4B	Transformer	In Service	In Service	Provides power for automatic transfer switch IT-4
64	10R	Panel	In Service	In Service	Contains power supplies for, and elements of, credited instrumentation
65	ER-622-080	Panel	In Service	In Service	Contains power supplies for, and elements of, credited instrumentation
66	ATS DC-D	Automatic Transfer Switch	In Service	In Service	Provides power for 125V DC distribution panel DC-D
67	6K3A	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
68	6K3B	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
69	6K5A	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
70	6K5B	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
71	6K4A	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
72	6K4B	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
73	6K6A	Isolation Condenser valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
74	6K6B	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	CR120AD0424 1AA relay
75	6K7A	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	27s Time Delay Drop-Out Relay Model 700RTC11200 U1
76	6K7B	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	27s Time Delay Drop-Out Relay Model 700RTC11200 U1
77	6K8A	Isolation Condenser Valve Hi Flow Isolation Logic	Energized	Energized	27s Time Delay Drop-Out Relay Model 700RTC11200 U1
78	6K8B	Isolation Condenser Valve Hi Flow isolation logic	Energized	Energized	27s Time Delay Drop-Out Relay Model 700RTC11200 U1
79	Y-6-42	Back-Up Air Supply Accumulator for Valve V-23-0013	Functional	Functional	Passive Component
80	Y-6-43	Back-Up Air Supply Accumulator for Valve V-23-0014	Functional	Functional	Passive Component
81	Y-6-44	Back-Up Air Supply Accumulator for Valve V-23-0015&16	Functional	Functional	Passive Component
82	V-6-953	Pilot Solenoid Air Supply Valve for V-23-0015	De-Energized	Energized	
83	V-6-954	Pilot Solenoid Air Supply Valve for V-23-0016	De-Energized	Energized	
84	V-6-902	Pilot Solenoid Air Supply Valve for V-23-0013	De-Energized	Energized	
85	V-6-903	Pilot Solenoid Air Supply Valve for V-23-0014	De-Energized	Energized	
86	V-6-950	Instrument Air Regulating Valve	Functional	Functional	McMaster-Carr Supply Co, 382M, Model: 4959K1
87	V-6-899	Instrument Air Regulating Valve	Functional	Functional	McMaster-Carr Supply Co, 382M, Model: 4959K1
88	V-6-898	Instrument Air Regulating Valve	Functional	Functional	Fisher Controls International LLC Model 67CFR-239
89	CIP-3	Continuous Instrument Panel No. 3	Energized	Energized	

Table A-1 Oyster Creek ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
90	ROTARY INVERTER	120V AC Supply for CIP-3 208/120V, 3PH, 4W	Energized	Energized	
91	12XR	Panel	In Service	In Service	Contains PNL-822-12XRCS1 Key lock bypass switch for purge valves, bypasses Isolation relays for hardened vent valves
92	IT-3	Automatic Transfer Switch	In Service	In Service	

Attachment B: Oyster Creek Reduced ESEL (low frequency items)

The reduced ESEL listed on the following table contains those items from Attachment A which are susceptible to damage from low frequency spectral accelerations, as defined in Section 2.2.1.1 of EPRI 3002000704 [2].

Table B-1 Oyster Creek Reduced ESEL

ESEL Item Number	Equipment		Operating State		Notes
	ID	Description	Normal State	Desired State	
43	T-39-2	Diesel Generator Fuel Oil Storage Tank	Standby	Standby	Passive Component

Attachment C: ESEL HCLPF Values and Failure Modes Tabulation

Table C-1 Oyster Creek ESEP HCLPF Values and Failure Mode Tabulation

ESEL Item Number	Equipment ID	Failure Mode	HCLPF (g)	Additional Discussion
43	T-39-2	Anchorage	0.53	HCLPF calculated in 14Q4241-CAL-002 [10]