UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2017

Mr. Bryan C. Hanson
President and Chief
Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 (CAC NOS. MF0823 AND MF0824)

Dear Mr. Hanson

On March 12, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events" and Order EA-12-051, "Order to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation," (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12054A736 and ML12054A679, respectively). The orders require holders of operating reactor licenses and construction permits issued under Title 10 of the *Code of Federal Regulations* Part 50 to modify the plants to provide additional capabilities and defense-in-depth for responding to beyond-design-basis external events, and to submit for review Overall Integrated Plans (OIPs) that describe how compliance with the requirements of Attachment 2 of each order will be achieved.

By letter dated February 28, 2013 (ADAMS Accession No. ML13060A126), Exelon Generation Company, LLC (Exelon, the licensee) submitted its OIP for Oyster Creek Nuclear Generating Station (OCNGS) in response to Order EA-12-049. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 19, 2014 (ADAMS Accession No. ML14030A513), and August 29, 2016 (ADAMS Accession No. ML16214A329), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated December 6, 2016 (ADAMS Accession No. ML16342C392), Exelon submitted a compliance letter and Final Integrated Plan (FIP) in response to Order EA-12-049. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-049.

By letter dated February 28, 2013 (ADAMS Accession No. ML13059A266), Exelon submitted its OIP for OCNGS in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated November 8, 2013 (ADAMS Accession No. ML13268A031), and August 29, 2016 (ADAMS Accession No. ML16214A329), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated December 1, 2016 (ADAMS Accession No. ML16336A441), OCNGS submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Exelon's strategies for OCNGS. The intent of the safety evaluation is to inform Exelon on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515-191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML15257A188). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact John Hughey, Orders Management Branch, Oyster Creek Project Manager, at 301-415-3204 or at John.Hughey@nrc.gov.

Sincerely,

John P. Boska

John P. Boska, Acting Chief
Orders Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket No.: 50-219
Enclosure:
Safety Evaluation

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RELATED TO ORDERS EA-12-049 AND EA-12-051

EXELON GENERATION COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

The earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant in March 2011 highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers already in place in nuclear power plants in the United States. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. During the events in Fukushima, the challenges faced by the operators were beyond any faced previously at a commercial nuclear reactor and beyond the anticipated design-basis of the plants. The U.S. Nuclear Regulatory Commission (NRC) determined that additional requirements needed to be imposed at U.S. commercial power reactors to mitigate such beyond-design-basis external events (BDBEEs).

On March 12, 2012, the NRC issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" [Reference 4]. This order directed licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities in the event of a BDBEE. Order EA-12-049 applies to all power reactor licensees and all holders of construction permits for power reactors.

On March 12, 2012, the NRC also issued Order EA-12-051, "Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation" [Reference 5]. This order directed licensees to install reliable SFP level instrumentation with a primary channel and a backup channel, and with independent power supplies that are independent of the plant alternating current (ac) and direct current (dc) power distribution systems. Order EA-12-051 applies to all power reactor licensees and all holders of construction permits for power reactors.

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC

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regulations and processes and determining if the agency should make additional improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 [Reference 1]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012, the NRC staff provided SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," [Reference 2] to the Commission. This paper included a proposal to order licensees to implement enhanced BDBEE mitigation strategies. As directed by the Commission in staff requirements memorandum (SRM)-SECY-12-0025 [Reference 3], the NRC staff issued Orders EA-12-049 and EA-12-051.

2.1 Order EA-12-049

Order EA-12-049, Attachment 2, [Reference 4] requires that operating power reactor licensees and construction permit holders use a three-phase approach for mitigating BDBEEs. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and SFP cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely. Specific requirements of the order are listed below:

1) Licensees or construction permit (CP) holders shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event.

2) These strategies must be capable of mitigating a simultaneous loss of all ac power and loss of normal access to the ultimate heat sink [UHS] and have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

3) Licensees or CP holders must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.

4) Licensees or CP holders must be capable of implementing the strategies in all modes of operation.

5) Full compliance shall include procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies.

On December 10, 2015, following submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, Revision 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," [Reference 6] to the NRC to provide revised specifications for an industry-developed methodology for the development, implementation, and maintenance of guidance and strategies in response to the Mitigation Strategies order. The NRC staff reviewed NEI 12-06, Revision 2, and on January 22, 2016, issued Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 1, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," [Reference 7], endorsing NEI 12-06, Revision 2, with exceptions, additions, and clarifications, as an acceptable means of meeting the requirements of Order EA-12-049, and published a notice of its availability in the *Federal Register* (81 FR 10283).

2.2 Order EA-12-051

Order EA-12-051, Attachment 2, [Reference 5] requires that operating power reactor licensees and construction permit holders install reliable SFP level instrumentation. Specific requirements of the order are listed below:

> All licensees identified in Attachment 1 to the order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.
>
> 1. The spent fuel pool level instrumentation shall include the following design features:
>
> 1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.
>
> 1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

1.4 Qualification: The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

1.5 Independence: The primary instrument channel shall be independent of the backup instrument channel.

1.6 Power supplies: Permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

1.7 Accuracy: The instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design shall provide for routine testing and calibration.

1.9 Display: Trained personnel shall be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display shall provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel shall be trained in the use and the provision of alternate power to the primary and backup instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels.

2.3 Testing and Calibration: Processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

On August 24, 2012, following several NEI submittals and discussions in public meetings with NRC staff, the NEI submitted document NEI 12-02, "Industry Guidance for Compliance With NRC Order EA-12-051, To Modify Licenses With Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1 [Reference 8] to the NRC to provide specifications for an industry-developed methodology for compliance with Order EA-12-051. On August 29, 2012, the NRC staff issued its final version of JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation" [Reference 9], endorsing NEI 12-02, Revision 1, as an acceptable means of meeting the requirements of Order EA-12-051 with certain clarifications and exceptions, and published a notice of its availability in the *Federal Register* (77 FR 55232).

3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 [Reference 10] Exelon Generation Company, LLC (Exelon, the licensee) submitted an Overall Integrated Plan (OIP) for Oyster Creek Nuclear Generating Station (OCNGS, Oyster Creek) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 28, 2014 [Reference 12], August 28, 2014 [Reference 13], February 27, 2015 [Reference 14], August 28, 2015 [Reference 38], February 26, 2016 [Reference 39], and August 26, 2016 [Reference 44] the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 15], the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 36]. By letters dated February 19, 2014 [Reference 16], and August 29, 2016 [Reference 17], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated December 6, 2016 [Reference 18], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP).

3.1 Overall Mitigation Strategy

Attachment 2 to Order EA-12-049 describes the three-phase approach required for mitigating BDBEEs in order to maintain or restore core cooling, containment, and SFP cooling capabilities. The phases consist of an initial phase (Phase 1) using installed equipment and resources, followed by a transition phase (Phase 2) in which portable onsite equipment is placed in service, and a final phase (Phase 3) in which offsite resources may be placed in service. The timing of when to transition to the next phase is determined by plant-specific analyses.

While the initiating event is undefined, it is assumed to result in an extended loss of ac power (ELAP) with a loss of normal access to the UHS. Thus, the ELAP with loss of normal access to the UHS is used as a surrogate for a BDBEE. The initial conditions and assumptions for the analyses are stated in NEI 12-06, Section 3.2.1, and include the following:

1. The reactor is assumed to have safely shut down with all rods inserted (subcritical).

2. The dc power supplied by the plant batteries is initially available, as is the ac power from inverters supplied by those batteries; however, over time the batteries may be depleted.
3. There is no core damage initially.
4. There is no assumption of any concurrent event.
5. Because the loss of ac power presupposes random failures of safety-related equipment (emergency power sources), there is no requirement to consider further random failures.

Oyster Creek is a General Electric boiling-water reactor (BWR) Model 2 with a Mark I containment. The licensee's three-phase approach to mitigate a postulated ELAP event, as described in the FIP, is summarized below. The approach is somewhat different if the plant receives warning of a pending flood, but the initial actions are similar.

At the onset of an ELAP the reactor is assumed to trip from full power. The main condenser becomes unavailable due to the automatic closure of the main steam isolation valves (MSIVs). Feed water flow to the reactor is lost and the Electromatic Relief Valves (EMRVs) automatically cycle to control pressure, causing reactor water level to decrease. Decay heat is initially removed by two isolation condensers (ICs). The IC valves open automatically and steam from the reactor pressure vessel (RPV) is cooled and condensed in the IC. The steam from the RPV condenses in the tube side of the IC and flows back to the RPV via natural circulation. The installed ICs provide core cooling and depressurization during the initial phase. Within 100 minutes of the ELAP, the licensee expects to establish makeup to the shell side of the ICs using a diesel driven FLEX pump. This time frame will ensure that the condenser tubes remain covered to maintain decay heat removal. The FLEX pump continues to provide makeup water to the ICs as needed to continue reactor cooling as well as depressurization which limits system leakage, thus reducing containment heatup. When the RPV pressure is sufficiently reduced, the FLEX pump provides injection into the RPV for cooling, depressurization, and inventory control.

The diesel driven FLEX pump takes water from the intake structure, which is supplied from Barnegat Bay, the UHS for the plant. The water from the UHS will be used to supply the shell side of the two ICs for an indefinite period of decay heat removal, provide makeup water to the RPV, and to fill or spray the spent fuel pool.

The reactor has a Mark I containment, which is inerted with nitrogen at power. The licensee performed a containment evaluation and determined that the containment pressure limits will not be exceeded throughout the event. The National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) is providing additional pumps in Phase 3 that can be used, if required, to provide water for containment cooling. Venting the primary containment to atmosphere is not anticipated.

The SFP is located in the reactor building. To maintain SFP cooling capabilities, the licensee plans to have available water makeup capability to the pool at six hours after the ELAP event and initiate makeup within 12 hours of the event. The pool will initially heat up due to the unavailability of the normal cooling system. The licensee has calculated that, depending on the spent fuel loading in the pool, boiling could start as soon as 10.3 hours after the start of the ELAP. The pool water level would drop to 10 ft. above the top of the fuel in approximately 45.6 hours. At this water level in the pool, access to the operating deck area could become compromised from a radiological perspective. The operators will deploy the hoses at the SFP deck before the onset of pool boiling. Habitable conditions on the operating deck are facilitated by propping open the reactor building doors and opening the roof hatch to allow vertical airflow as building temperatures rise. These actions are to be performed prior to the spent fuel pool boiling.

The licensee has a primary and alternate strategy to provide makeup water to the pool. The primary SFP strategy is to supply water from the intake canal using the FLEX pump with hoses connected to the SFP diffuser piping. The alternate strategy is to route a hose directly into the SFP at the SFP deck area or connect it to a nozzle that sprays the water over the spent fuel. The NSRC is providing additional pumps in Phase 3 that can be used as backup if required to provide water for spent fuel pool cooling.

In order to extend the safety-related battery life, the operators will complete dc bus load stripping within the initial 90 minutes following the ELAP event. A 500-kilowatt (kW), 480 volt alternating current (Vac) FLEX generator will be deployed from a FLEX storage pad and will enable re-energizing existing station battery chargers and vital instrumentation within 2.5 hours, which is well inside the battery coping times with or without dc load shed.

In addition, a NSRC will provide high capacity pumps and large diesel-driven generators (DGs), which could be used to as backup to the FLEX equipment used in Phase 2. There are two NSRCs in the United States.

Below are specific details on the licensee's strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of a BDBEE, and the results of the staff's review of these strategies. The NRC staff evaluated the licensee's strategies against the endorsed NEI 12-06, Revision 2, guidance.

3.2 Reactor Core Cooling Strategies

Order EA-12-049 requires licensees to maintain or restore cooling to the reactor core in the event of an ELAP concurrent with a loss of normal access to the UHS. Although the ELAP results in an immediate trip of the reactor, sufficient core cooling must be provided to account for fission product decay and other sources of residual heat. Consistent with endorsed guidance from NEI 12-06, Phase 1 of the licensee's core cooling strategy credits installed equipment (other than that presumed lost to the ELAP with loss of normal access to the UHS) that is robust in accordance with the guidance in NEI 12-06. In Phase 2, robust installed equipment is supplemented by onsite FLEX equipment, which is used to cool the core either directly (e.g., pumps and hoses) or indirectly (e.g., FLEX electrical generators and cables repowering robust installed equipment). The equipment available onsite for Phases 1 and 2 is further supplemented in Phase 3 by equipment transported from the NSRCs.

As reviewed in this section, the licensee's core cooling analysis for the ELAP with loss of normal access to the UHS event presumes that, per endorsed guidance from NEI 12-06, the unit would have been operating at full power prior to the event. Therefore, the suppression pool may be credited as the heat sink for core cooling during the ELAP with loss of normal access to the UHS event. Maintenance of sufficient RPV inventory, despite steam release from the SRVs and ongoing system leakage expected under ELAP conditions, is accomplished through a combination of installed systems and FLEX equipment. The specific means used by the licensee to accomplish adequate core cooling during the ELAP with loss of normal access to the

UHS event are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions where the unit is shut down or being refueled is reviewed separately in Section 3.11 of this evaluation.

3.2.1 Core Cooling Strategy and RPV Makeup

3.2.1.1 Phase 1

As a result of the loss of electrical power that initiates the ELAP event, the reactor is assumed to trip, with all control rods inserting into the core. The loss of electrical power would further result in isolation of the reactor via the closure of valves on piping (including the main steam and feedwater lines) that connects the reactor vessel to equipment located outside of the containment building. Although the worth of the inserted control rods should be sufficient to maintain the reactor core subcritical throughout the ELAP event, fission product decay and other sources of residual heat continue to heat up and pressurize the primary system following the reactor trip.

With the reactor isolated from the main feedwater system and main condenser, the licensee's Phase 1 strategy for core cooling relies upon the use of installed plant equipment, including the two ICs and EMRVs, to remove residual heat and decay heat from the reactor. The ICs are placed in service when the dc powered condensate return valves open on either a high reactor pressure signal or a low-low reactor water level signal from the Reactor Protection System; at Oyster Creek, this system is credited to operate automatically under analyzed ELAP conditions, with the EMRVs discharging steam from the reactor vessel in response to elevated reactor pressure. However, in accordance with plant emergency procedures, operators would assume manual control of the equipment necessary to provide core cooling. Operators would establish an initial core cooldown rate of 10 degrees Fahrenheit/hour (°F/hr) or less, in accordance with Abnormal Operating Procedures ABN-1, "Reactor Scram", and ABN-36, "Loss of Offsite Power & Station Blackout". When operators recognize that a potential ELAP event is in progress, ABN-36 directs increasing the cooldown rate to 50 °F/hr. The licensee's FIP states that this recognition would be made no later than 10 minutes into the event.

The ICs at Oyster Creek are designed to operate for 100 minutes without makeup to the shell side of the condensers, when both ICs are in operation. Therefore, at or before 100 minutes into the ELAP event, portable FLEX equipment must be placed in service to continue core cooling using the ICs. The licensee's thermal-hydraulic analysis concludes that if shell-side makeup is established to the ICs prior to 100 minutes into the ELAP event, water level in the RPV will remain above the top of active fuel (TAF) for at least 3.3 hours from the time of the initial event. No makeup to the RPV is required as part of the Phase 1 strategy.

3.2.1.2 Phase 2

In Phase 2, operators will deploy a portable, diesel-driven, centrifugal FLEX pump rated at 1100 gallons per minute (gpm) at 150 pound per square inch differential (psid). The discharge of this FLEX pump will be aligned to a FLEX distribution manifold, which in turn would discharge simultaneously to the RPV, spent fuel pool, and the shell side of the ICs. This FLEX distribution manifold is to be staged in the Reactor Building at the 23-ft. elevation. Operators will align suction to the FLEX pump from a chemistry sample point in the circulating water discharge tunnel, which establishes an essentially unlimited supply of makeup water from the intake/discharge canal, which draws from the UHS (Barnegat Bay). An alternate suction point is available at the intake tunnel, at the area between the intake traveling screens and the circulating water pumps. Both the discharge tunnel and the intake tunnel draw from the single intake/discharge canal, and ultimately Barnegat Bay.

The primary flowpath from the FLEX pump discharge to the ICs is from the 23-ft. FLEX manifold to a second FLEX manifold on the 95-ft. level of the reactor building, and then to a common drain connection of the two ICs. An alternate flowpath can be established, if necessary, via a hose from the 23-ft. FLEX manifold to a different connection on condensate transfer system piping which connects to the shell sides of both ICs.

Deployment of the FLEX pump and its associated hoses will begin immediately upon operator recognition that an ELAP is in progress, no later than 10 minutes into the event. The licensee has validated that the pump will be aligned and ready to provide makeup to the ICs no later than 90 minutes into the ELAP event; therefore, use of the ICs for core cooling will not be interrupted by low water level in the shell side of the condensers. Operators would continue cooling the reactor at a rate of 50 °F/hr until the RPV has cooled down to the point that heat transfer to the ICs is no longer effective.

The FLEX pump will be aligned to provide makeup to the RPV via the core spray system within 3.3 hours of the initial event, which will maintain RPV level above the TAF. The primary RPV makeup connection is on Core Spray System 1; the alternate flowpath is via Core Spray System 2. These two flowpaths are independent from each other, which meets the guidance of NEI 12-06.

The licensee's FLEX strategy for core cooling involves the use of raw water sources. The licensee's strategy attempts to minimize the potential for inadequate core cooling as a result of water-quality issues via a suction strainer on the FLEX pump, which serves to protect the pump from debris, as well as to prevent blockage in downstream flowpaths. This suction strainer is designed with holes no larger than a diameter of 15/16 in., compared with the 1.049 in. inner diameter of the 1 in. schedule 40 elbows of the core spray system. These elbows are the most limiting component in either of the RPV and IC makeup flowpaths. Therefore, the FLEX pump suction strainer will effectively prevent any debris from the raw water source from impairing the RPV or IC makeup flowpaths. In addition, the primary and alternate suction points for the FLEX pump are located downstream of the intake screens of the intake/discharge canal. FLEX Support Guideline (FSG) FSG-00, FLEX Strategy Implementation, directs operators to establish a clean source of makeup water to the ICs and RPV when possible, either by aligning a surviving on-site makeup water tank, if available (e.g. the condensate storage tank (CST)) or by procuring a salt water filtration system. Further, the staff notes that the licensee plans to inject to the RPV via the core spray system, which reduces the potential for the use of raw water to cause core inlet blockage that could impede core cooling.

3.2.1.3 Phase 3

The Phase 3 strategy for core cooling at Oyster Creek is a continuation of the Phase 2 strategy, supplemented by portable pumps and generators furnished by the NSRC. The licensee's FIP states that the initial delivery of Phase 3 equipment to the site will occur within 24 hours of notification of an ELAP event. The Phase 3 equipment would serve as backup or redundant equipment to the on-site Phase 2 portable equipment; if necessary, the Phase 2 mechanical and electrical connections would still be used when placing the NSRC equipment in service.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

The limiting flood event for Oyster Creek is a local intense precipitation (LIP) event. The Flood Hazard Reevaluation Report states that the reevaluated LIP event is the only flood-causing event that exceeds the current design basis flood; the flood elevation height increased above the current license basis value (23.5 ft.) by 0.88 ft.. Accordingly, the licensee has incorporated protective measures against flooding into the Oyster Creek FLEX strategy. The licensee's FIP concludes that the reevaluated flood hazard will not prevent the implementation of the FLEX strategy, provided these specific flooding protective measures are put in place. Section 3.5.2 of this safety evaluation (SE) contains more information about the potential flooding hazard at Oyster Creek.

3.2.3 Staff Evaluations

3.2.3.1 Availability of Structures, Systems, and Components (SSCs)

Guidance document NEI 12-06 provides guidance that the baseline assumptions have been established on the presumption that other than the loss of the ac power sources and normal access to the UHS, installed equipment that is designed to be robust with respect to design-basis external events is assumed to be fully available. Installed equipment that is not robust is assumed to be unavailable. Below are the baseline assumptions for the availability of SSCs for core cooling during an ELAP caused by a BDBEE.

3.2.3.1.1 Plant SSCs

Isolation Condensers

The licensee explained that the ICs are designed to remove 410 X 10^6 Btu/hr. and can provide emergency cooling for 100 minutes without shell-side makeup. In addition, the reactor water level will remain above the top of active fuel for at least 3.3 hours with shell-side makeup made available to the ICs prior to 100 minutes from the initial time of the event. After 100 minutes, make up to the ICs shells is necessary to keep the shell-side level above the tube bundles; however, once the FLEX Pump provides makeup to the ICs with suction from the Intake/Discharge Canal, the system will be capable of indefinite operation. Updated Final Safety Analysis Report (UFSAR) Table 3.2-1 indicates that the IC System was designed to be Seismic Class I and is located within the Reactor Building, which is Class I structure protected from externally generated missiles. Based on the design and location of the IC system within the Reactor Building, the staff finds this system is robust and is expected to be available during an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

3.2.3.1.2 Plant Instrumentation

The licensee's FIP identifies the following instrumentation required to implement the core cooling and decay heat removal strategy:

- RPV level (channels C and D)

- RPV wide range pressure (channels C and D)
- IC shell level (channels A and B)

This instrumentation is consistent with or in excess of the recommendations specified in the endorsed guidance of NEI 12-06.

The licensee plans to monitor indication for the above instruments from the main control room. This instrumentation is powered by batteries and will be maintained throughout the event; it will be available prior to and after load stripping of the dc buses during Phase 1, and supported via battery chargers powered by the FLEX DGs in Phases 2 and 3. Therefore, based upon the information provided by the licensee, the NRC staff understands that indication for the above instruments would be available and accessible continuously throughout the ELAP event.

The licensee's procedures identify specific locations for taking local readings and provide factors for converting the measured signal into the appropriate measurement units. The use of the instruments is described in FSG-20, "Alternate Instrumentation Reading". Procedure FSG-20 contains procedures for obtaining indications for RPV water level and pressure, suppression pool level and temperature, and containment pressure.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee's mitigating strategy for reactor core cooling is based in part on thermal-hydraulic analysis performed using Version 4 of the Modular Accident Analysis Program (MAAP). Because the thermal-hydraulic analysis for the reactor core and containment during an ELAP event are closely intertwined, as is typical of BWRs, Oyster Creek has addressed both in a single, coupled calculation. This dependency notwithstanding, the NRC staff's discussion in this section of the SE focuses on the licensee's analysis of reactor core cooling. The NRC staff's review of the licensee's analysis of containment thermal-hydraulic behavior is provided subsequently in Section 3.4.4.2 of this evaluation.

MAAP is an industry-developed, general-purpose thermal-hydraulic computer code that has been used to simulate the progression of a variety of light water reactor accident sequences, including severe accidents, such as the Fukushima Dai-ichi event. Initial code development began in the early 1980s, with the objective of supporting an improved understanding of and predictive capability for severe accidents involving core overheating and degradation in the wake of the accident at Three Mile Island Nuclear Station, Unit 2. Currently, maintenance and development of the code is carried out under the direction of the Electric Power Research Institute (EPRI).

To provide analytical justification for their mitigating strategies in response to Order EA-12-049, a number of licensees for BWRs and pressurized-water reactors (PWRs) completed analysis of the ELAP event using Version 4 of the MAAP code (MAAP4). Although MAAP4 and predecessor code versions have been used by industry for a range of applications, such as the analysis of severe accident scenarios and probabilistic risk analysis (PRA) evaluations, the NRC staff had not previously examined the code's technical adequacy for performing best-estimate simulations of the ELAP event. In particular, due to the breadth and complexity of the physical phenomena within the code's calculation domain, as well as its intended capability for rapidly simulating a variety of accident scenarios to support PRA evaluations, the NRC staff observed that the MAAP code makes use of a number of simplified correlations and approximations that should be evaluated for their applicability to the ELAP event. Therefore, in support of the staff's reviews of licensees' strategies for ELAP mitigation, the NRC staff audited the capability of the MAAP4 code for performing thermal-hydraulic analysis of the ELAP event for both BWRs and PWRs. The NRC staff's audit review involved a limited review of key code models, as well as confirmatory analysis with the TRACE code to obtain an independent assessment of the predictions of the MAAP4 code.

To support the NRC staff's review of the use of MAAP4 for ELAP analyses, in June 2013, EPRI issued a technical report entitled "Use of Modular Accident Analysis Program (MAAP) in Support of Post-Fukushima Applications." The document provided general information concerning the code and its development, as well as an overview of its physical models, modeling guidelines, validation, and quality assurance procedures.

Based on the NRC staff's review of EPRI's June 2013 technical report, as supplemented by further discussion with the code vendor, audit review of key sections of the MAAP code documentation, and confirmation of acceptable agreement with NRC staff simulations using the TRACE code, the NRC staff concluded that, under certain conditions, the MAAP4 code may be used for best-estimate prediction of the ELAP event sequence for BWRs. The NRC staff issued an endorsement letter dated October 3, 2013, which documented these conclusions and identified specific limitations that BWR licensees should address to justify the applicability of simulations using the MAAP4 code for demonstrating that the requirements of Order EA-12-049 have been satisfied.

The licensee addressed the limitations from the NRC staff's endorsement letter in its 6-month status reports to the NRC. The licensee's response utilized the generic roadmap document and response template that had been developed by EPRI. The NRC staff's audit review of this information, as well as its audit of the OCNGS plant-specific MAAP analysis, confirmed that the licensee had acceptably addressed the limitations from the endorsement letter as follows:

- In addressing the NRC staff's request for providing benchmarks supporting the MAAP code's applicability to Oyster Creek, the licensee cited the generic BWR roadmap document prepared by EPRI. The NRC staff's review of EPRI's BWR roadmap document determined that sufficient benchmark comparisons had been made to support application of the MAAP code to the beyond-design-basis ELAP event for BWRs.

- The NRC staff requested that the collapsed level in the reactor vessel remain above the top of the active fuel region and that the cooldown rate remain within the technical specification (TS) limits. The results from the licensee's representative simulation with the MAAP code (Case 7 of the analysis) indicated that the limitation on water level should be satisfied for the analyzed ELAP event, as the collapsed level in the reactor vessel would remain above the top of active fuel region for the duration of the analysis. Regarding the limitation concerning the cooldown rate, as discussed in Section 3.2.1.1 of this evaluation, operators would cool down the RPV at a rate of 50 °F/hr, which is within the TS limit of 100 °F/hr.

- In addressing the NRC staff's request that the licensee's use of MAAP should be consistent with certain sections of a June 2013 Nuclear Energy Institute (NEI) position paper on the use of MAAP, the licensee confirmed that the position paper had been followed in the analysis performed for Oyster Creek. The NRC staff's audit did not identify inconsistencies between the NEI position paper and the licensee's analysis.

- The NRC staff requested that the licensee identify key modeling parameters used in the analysis for the ELAP event and provide justification for their adequacy. The licensee responded according to a generic industry template. The NRC staff's audit did not identify any issues with the parameters assumed by the licensee and confirmed in particular that appropriate inputs and modeling options had been selected for the code parameters expected to have dominant influence for the ELAP event.

- In response to the NRC staff's request, the licensee's MAAP analysis was made available to the NRC staff during the audit. An overview of the NRC staff's audit review is noted below.

The licensee's original MAAP analysis (OC-MISC-010, Revision 0) considered six scenarios with varying timelines for plant cooldown rate, containment venting, and RPV injection. All six cases assumed that the reactor had operated at 100 percent power for 100 days at the time of event initiation, that all rods inserted at the scram, that the CST was unavailable, and that ICs started automatically. Additionally, all cases assumed a 35 gpm reactor coolant system (RCS) leakage rate, which includes recirculation pump seal leakage, identified and unidentified RCS leakage. This leakage rate was conservatively assumed to begin immediately upon the initiating event, and would gradually decrease as RPV pressure decreased. This value for RCS leakage rate is further discussed in Section 3.2.3.3 of this SE.

Originally, the licensee selected Case 6 as the representative case for modeling the actual FLEX strategy; Case 6 assumed a 50 °F/hr cooldown rate and FLEX RPV injection beginning at 3.8 hours into the event. The analysis concluded that for this run, no failure conditions were met (e.g. exceeding the Pressure Suppression Pressure curve or Heat Capacity Temperature Limit curve) and containment venting would not be required. Another conclusion was that collapsed RPV water level would briefly drop below the TAF, remaining near that level for a few minutes before RPV injection caused collapsed water level to rapidly increase.

To ensure that collapsed water level would be prevented from dropping below TAF, Revision 1 of the MAAP analysis added a seventh test run (Case 7), which initiated FLEX RPV injection earlier (at 3.3 hours into the event, as opposed to 3.8 hours) and delayed simultaneous operation of the ICs until RPV water level had recovered. Case 7 of the revised analysis demonstrated that containment venting would not be required for the duration of the ELAP event, and that collapsed RPV water level would remain above TAF throughout the event. Therefore, based on the evaluation above, the NRC staff concludes that the licensee's analytical approach should appropriately determine the sequence of events for reactor core cooling, including time-sensitive operator actions, and evaluate the required equipment to mitigate the analyzed ELAP event, including pump sizing and cooling water capacity.

3.2.3.3 Recirculation Pump Seals

An ELAP event would result in the interruption of cooling to the recirculation pump seals, potentially resulting in increased leakage due to the distortion or failure of the seals, elastomeric O-rings, or other components. Sufficient primary make-up must be provided to offset recirculation pump seal leakage and other expected sources of primary leakage, in addition to removing decay heat from the reactor core.

Oyster Creek has CAN2A recirculation pump seals. The licensee's calculations assume an initial seal leakage rate at full system pressure and temperature of 20 gpm total for all five recirculation pumps, which contributes to a total assumed RCS leakage rate of 35 gpm. The seals are credited to not fail catastrophically during the ELAP.

During the audit, the NRC staff discussed recirculation pump seal leakage with the licensee and requested that the licensee justify the applicability of the assumed leakage rate to the ELAP event. The licensee has Model CAN2A recirculation pump seals that were manufactured by Atomic Energy of Canada Limited. The first CAN2A seal was installed at a U.S. BWR in 1986 to provide improved overall seal performance, with consideration given to postulated station blackout (SBO) conditions wherein seal cooling is lost. BWRs such as Oyster Creek and Nine Mile Point Unit 1 which use CAN2A seals rely on ICs in lieu of a turbine-driven pump for cooling an isolated reactor core. SBO testing was performed in the early 1990s with relatively strict acceptance criteria. The results of the CAN2A station blackout testing showed that, provided that conditions are maintained within a qualification envelope, the seal faces did not "pop open," nor did the seals otherwise experience excessive leakage when seal cooling was lost.

Considering the above factors, the NRC staff concludes that the leakage rate assumed by the licensee is reasonable based on work performed in support of the Generic Letter 91-07, "GI [Generic Issue] -23, 'Reactor Coolant Pump Seal Failures' and Its Possible Effect on Station Blackout." The staff further notes that gross seal failures are not anticipated to occur during the postulated ELAP event. As has been noted above, unlike the majority of U.S. BWRs, OCNGS does not have an installed steam-driven pump (i.e., RCIC [reactor core isolation cooling]) capable of injecting into the primary system at a flow rate in excess of the primary system leakage rate expected during an ELAP. However, the licensee has demonstrated using MAAP4 analysis (see Section 3.2.3.2 of this SE) that RPV water level will remain above TAF throughout the event, provided that the FLEX pump is aligned to inject into the RPV by 3.3 hours into the event. The RPV makeup rate from the FLEX pump (100 gpm) is more than double the assumed seal leakage rate, and therefore should have the capacity to overcome the losses and increase the RPV level.

Based upon the discussion above, the NRC staff concludes that the recirculation pump seal leakage rates assumed in the licensee's thermal-hydraulic analysis may be applied to the beyond-design-basis ELAP event for the site.

3.2.3.4 Shutdown Margin Analyses

As described in its UFSAR, the Oyster Creek reactor design is such that the control rods provide adequate shutdown margin under all anticipated plant conditions, with the assumption that the highest-worth control rod remains fully withdrawn. Additional margin is added to account for the expected impacts of burnup during the operating cycle. The OCNGS TSs further clarify that shutdown margin is to be calculated for a cold, xenon-free condition to ensure that the most reactive core conditions are bounded.

Based on the NRC staff's audit review, the licensee's ELAP mitigating strategy maintains the reactor within the envelope of conditions analyzed by the licensee's existing shutdown margin calculation. Furthermore, the existing calculation retains conservatism because the guidance in NEI 12-06 permits analyses of the beyond-design-basis ELAP event to assume that all control rods fully insert into the reactor core, whereas, the licensee demonstrated adequate shutdown margin using the licensing-basis assumption of the strongest rod being fully withdrawn.

The licensee stated that for Oyster Creek Cycle 25, a minimum shutdown margin of 1.39 percent Δk/k would be available, approximately 3-4 times the minimum requirement in TSs. The licensee further stated that justification for adequate shutdown margin for the ELAP event will be verified for future operating cycles.

Therefore, based on the evaluation above, the NRC staff concludes that the sequence of events in the proposed mitigating strategy should result in acceptable shutdown margin for the analyzed ELAP event.

3.2.3.5 FLEX Pumps and Water Supplies

In the FIP, Section 2.3.9 states that an RPV makeup flowrate of 100 gpm was assumed with an assumed constant RPV leakage rate of 35 gpm despite slowly lowering RPV pressure. Furthermore, the 100 gpm (35 gpm + margin) RPV makeup flowrate is a conservative assumption to ensure the RPV water level remains above TAF and begins to rise if injection begins 3.3 hours from event onset. During its audit, the staff reviewed the licensee's hydraulic calculation and noted that the FLEX Pump can deliver a minimum makeup rate of 235 gpm at 150 pounds per square inch gage (psig) at the time RPV makeup is required to ensure level is above TAF. The staff noted the hydraulic calculation demonstrates for the bounding scenario that the FLEX Pump with suction from the intake/discharge canal water from Barnegat Bay can simultaneously supply makeup to the necessary users (i.e., IC, RPV and/or SFP) and still provide a margin of 135 gpm for RPV makeup.

In the FIP, Section 2.3.9 states that an IC makeup flowrate of 187 gpm is necessary to remove reactor decay heat at the start of the event. In addition, the required makeup will decrease as the ELAP event progresses when decay heat decreases over time. The licensee explained that the makeup method will be to use batch feeding (opening and closing the makeup valve) rather than by attempting to match makeup flow to inventory used. The hydraulic calculation determined that with the IC makeup valve full open, the flow to the ICs at the time makeup is required (i.e., ~1.5 hours after the initiating event) is 424 gpm at 121 psig, which is significantly greater than the minimum required flowrate. During its audit, the staff reviewed the licensee's hydraulic calculation and noted that the FLEX Pump can deliver a minimum makeup rate of 237 gpm at 52 psig approximately 12 hours after the initiating event. The staff noted the hydraulic calculation demonstrates for the bounding scenario that the FLEX Pump with suction from the intake/discharge canal water from Barnegat Bay can simultaneously supply makeup to the necessary users (i.e., IC, RPV and/or SFP) and still provide a margin of 50 gpm for IC makeup.

As described in the FIP, the RPV and IC makeup strategy relies on a single FLEX Pump to provide makeup water during Phase 2. A total of two FLEX Pumps are stored on the FLEX

Storage Pads (i.e., northwest and southeast quadrants of the site inside the protected area. FIP Section 2.3.9 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX Pump. Specifically the pump is trailer mounted Dri-Prime HL 130M Godwin Pump (rated at approximately 1100 gpm at 150 psid) and is a self-priming, variable speed, diesel engine driven centrifugal pump. The staff noted that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3, as described in FIP Table 5, would allow the NSRC pump to fulfill the mission of the onsite FLEX pump if the onsite FLEX pump were to fail.

The staff confirmed that flow rates and pressures evaluated in the hydraulic analyses were reflected in the FIP for the respective RPV and IC makeup strategies based upon the above FLEX pumps being diesel driven and respective FLEX connections being made as directed by the FSGs. During the onsite audit, the staff conducted a walk down of the hose deployment routes for the above FLEX pumps to confirm the evaluations of the pump staging locations, hose distance runs, and connection points as described in the above hydraulic analyses and FIP.

Based on the staff's review of the FLEX pumping capabilities, as described in the above hydraulic analyses and the FIP, the licensee has demonstrated that its FLEX pump should perform as intended to support RPV makeup and IC makeup during an ELAP event, consistent with NEI 12-06, Section 11.2.

3.2.3.6 Electrical Analyses

The licensee's electrical strategies provide power to the equipment and instrumentation used to mitigate the ELAP and LUHS. The electrical strategies described in the FIP are practically identical for maintaining or restoring core cooling, containment, and SFP cooling, except as noted in Sections 3.3.4.4 and 3.4.4.4 of this SE.

The NRC staff reviewed the licensee's FIP, conceptual electrical single-line diagrams, and the summary of calculations for sizing the FLEX generators and station batteries. The staff also reviewed the licensee's evaluations that addressed the effects of temperature on the electrical equipment credited in the FIP as a result of the loss of heating, ventilation, and air conditioning (HVAC) caused by the event.

According to the licensee's FIP, operators would declare an ELAP following a loss of offsite power, emergency diesel generators (EDGs) and gas turbine generators. The plants indefinite coping capability is attained through the implementation of pre-determined FLEX strategies that are focused on maintaining or restoring key plant safety functions. A safety function-based approach provides consistency with, and allows coordination with, existing plant emergency operating procedures (EOPs). The FLEX strategies are implemented in support of EOPs using FSGs.

During the first phase of the ELAP event, Oyster Creek would rely on the Class 1E station batteries to provide power to key instrumentation for monitoring parameters and power to controls for SSCs used to maintain the key safety functions (Reactor core cooling, RCS inventory control, and Containment integrity). The Oyster Creek Class 1E station batteries and associated dc distribution systems are located in the reactor building, which is a safety-related structure designed to meet all applicable design-basis external hazards. The licensee's procedure FSG-04, “FLEX Support Guide 4 DC Load Shed,” directs operators to conserve dc power during the event by stripping non-essential loads. Operators will strip or shed unnecessary loads to extend battery life until backup power is available. The plant operators would commence load shedding within 10 minutes and complete load shedding within 90 minutes from the onset of an ELAP/LUHS event.

Oyster Creek has two Class 1E 125 Volt (V) dc station batteries (B and C). Station batteries B and C are manufactured by Lucent (now Tyco) and Exide Technologies, respectively. Station battery B is model AT&T Round Cell KS 20472L with a capacity of 1600 ampere-hours (A-H) at an 8-hour discharge rate to 1.75 V per cell. Station battery C is model GNB, NCN-1200 with a capacity of 1200 A-H at an 8-hour discharge rate to 1.75 V. The licensee noted and the staff confirmed that the Battery B and Battery C capacity could be extended to 18 hours and 25 hours of coping, respectively, until station battery chargers are returned to service.

The NEI White Paper, "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13241A186) provides guidance for calculating extended duty cycles of batteries (i.e., beyond 8 hours). This paper was endorsed by the NRC (ADAMS Accession No. ML13241A188). In addition to the White Paper, the NRC sponsored testing at Brookhaven National Laboratory that resulted in the issuance of NUREG/CR-7188, "Testing to Evaluate Extended battery Operation in Nuclear Power Plants," in May of 2015. The testing provided additional validation that the NEI White Paper method was technically acceptable. The NRC staff reviewed the licensee's battery calculations and confirmed that they had followed the guidance in the NEI White Paper.

The NRC staff reviewed the licensee's Nexus Report 12-4159.OCNGS, "Exelon Corporation Oyster Creek Generating Station (OCNGS) Battery Coping Evaluation Report," Revision 0, which verified the capability of the dc system to supply power to the required loads during the first phase of the Oyster Creek FLEX mitigation strategy plan for an ELAP as a result of a BDBEE. The licensee's evaluation identified the required loads and their associated ratings (ampere (A) and minimum required voltage) and the non-essential loads that would be shed within 90 minutes to ensure battery B and C operation can be extended from 5.85 to 18 hours and 14.28 to 25 hours, respectively.

Based on its review of the licensee's analysis and procedures, the battery vendor's capacity and discharge rates for the Class 1E station batteries, the NRC staff finds that the Oyster Creek dc systems have adequate capacity and capability to power the loads required to mitigate the consequences during Phase 1 of an ELAP as a result of a BDBEE provided that necessary load shedding is completed within the times assumed in the licensee's analysis.

The licensee's Phase 2 strategy includes repowering 480 Vac buses within 2.5 hours after initiation of an ELAP event using a portable 500-kW 480 Vac FLEX DG. The FLEX DG would supply power to the OCNGS 480 Vac vital bus (unit substation (USS) 1A2 or 1B2) circuits providing continuity of key parameter monitoring and other required loads. The FLEX DG would provide power to loads such as the vital battery chargers, recirculation loop valves, post-accident instrument panels, lighting, a control rod drive pump and a battery room exhaust fan.

The NRC staff reviewed the licensee's action request (A/R) 2386806, "Fukushima FLEX Electrical Mitigation Strategy Tech Eval," dated 10/30/2015, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the EDGs. Based on the NRC staff's review, the required loads for the Phase 2 500 kW FLEX DGs is about 398 kW. Therefore, one 500 kW FLEX DG is adequate to support the electrical loads required for the licensee's Phase 2 strategy.

If the "N" FLEX DG becomes unavailable or is out of service for maintenance, the other ("N+1") FLEX DG would be deployed to continue to support the required loads. The "N+1" FLEX DG is identical to the "N" FLEX DG, thus ensuring electrical compatibility and sufficient electrical capacity in an instance where substitution is required. Since the "N+1" FLEX DG is identical and interchangeable with the "N" FLEX DG, the NRC staff finds that the licensee has met the provisions of NEI 12-06, for spare equipment capability regarding the Phase 2 FLEX DGs.

For Phase 3, the licensee plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources if necessary. The offsite resources that will be provided by an NSRC includes two 1-megawatt (MW) 4160 Vac combustion turbine generators (CTGs), one 1100 kW 480 Vac CTG, and distribution panels (including cables and connectors). The licensee plans to only connect the 480 Vac CTGs and not the 4160 Vac CTGs. Based on the additional margin available due to the higher capacity (1100 kW) of the 480 Vac CTGs as compared to the Phase 2 FLEX DGs (500 kW), the NRC staff finds that the 480 Vac CTGs being supplied from an NSRC has sufficient capacity and capability to supply the required loads.

Based on its review, the NRC staff finds that the plant batteries used in the strategy should have sufficient capacity to support the licensee's strategy, and that the FLEX DGs and turbine generators that the licensee plans to use should have sufficient capacity and capability to supply the necessary loads during an ELAP event.

3.2.4 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that should maintain or restore core cooling and RCS inventory during an ELAP event consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-1 and Appendix C summarize an approach consisting of two separate capabilities for the SFP cooling strategies. This approach uses a portable injection source to provide the capability for 1) makeup via hoses on the refueling floor capable of exceeding the boil-off rate for the design-basis heat load; and 2) makeup via connection to spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load. However, in JLD-ISG-2012-01, Revision 1 [Reference 7], the NRC staff did not fully accept this approach, and added another requirement to either have the capability to provide spray flow to the SFP, or complete an SFP integrity evaluation which demonstrates that a seismic event would have a very low probability of inducing a crack in the SFP or its piping systems so that spray would not be needed to cool the spent fuel. The evaluation must use the reevaluated seismic hazard described in Section 3.5.1 below if it is higher than the site's current SSE. During the event, the licensee selects the SFP makeup method to use based on plant conditions. This approach also requires a strategy to mitigate the effects of steam from the SFP, such as venting.

As described in NEI 12-06, Section 3.2.1.7, and JLD-ISG-2012-01, Section 2.1, strategies that must be completed within a certain period of time should be identified and a basis that the time can be reasonably met should be provided. NEI 12-06, Section 3, provides the performance attributes, general criteria, and baseline assumptions to be used in developing the technical basis for the time constraints. Since the event is beyond-design-basis, the analysis used to provide the technical basis for time constraints for the mitigation strategies may use nominal initial values (without uncertainties) for plant parameters, and best-estimate physics data. All equipment used for consequence mitigation may be assumed to operate at nominal set points and capacities. In NEI 12-06, Section 3.2.1.2 describes the initial plant conditions for the at-power mode of operation; Section 3.2.1.3 describes the initial conditions; and Section 3.2.1.6 describes SFP initial conditions.

In NEI 12-06, Section 3.2.1.1 provides the acceptance criterion for the analyses serving as the technical basis for establishing the time constraints for the baseline coping capabilities to maintain SFP cooling. This criterion is keeping the fuel in the SFP covered with water.

The ELAP causes a loss of cooling in the SFP. As a result, the pool water will heat up and eventually boil off. The licensee's response is to provide makeup water. The timing of operator actions and the required makeup rates depend on the decay heat level of the fuel assemblies in the SFP. The sections below address the response during operating, pre-fuel transfer or post-fuel transfer operations. The effects of an ELAP with full core offload to the SFP is addressed in Section 3.11. The licensee decided to provide the spray flow described in JLD-ISG-2012-01.

3.3.1 Phase 1

In the FIP, Section 2.5.1 states that the water in SFP will gradually heat up and evaporate from decay heat following a loss of SFP cooling after the ELAP event occurs. Adequate SFP inventory exists to provide radiation shielding for personnel well beyond the time of boiling. During Phase 1, operators will establish ventilation pathways in the Reactor Building, deploy hoses in the Reactor Building to support SFP FLEX strategies and monitor level using the SFP level instrumentation installed per Order EA-12-051. The licensee explained that within the first 6 hours, a FLEX Pump and necessary hoses for SFP makeup will be physically staged and available, as necessary, to restore and maintain the normal level in Phase 2, even though the FLEX strategy shows SFP makeup starting at 12 hours after the initiating event.

3.3.2 Phase 2

During Phase 2, FIP Section 2.5.2 states that operators will deploy a portable FLEX Pump to supply water from the intake/discharge canal water from Barnegat Bay to the SFP. The FLEX Pump discharge can be routed to a connection to the SFP diffuser return line connection on the Reactor Building 75-ft. elevation (not requiring refueling floor access), or routed to the refuel floor to provide direct makeup and/or spray to the pool via portable oscillating spray nozzles on the Reactor Building 119-ft. elevation. The licensee's sequence of events indicates that 12 hours after the initiating event, the ability to provide makeup/spray to the SFP using the FLEX Pump will begin.

3.3.3 Phase 3

In the FIP, Section 2.5.3 indicates that SFP cooling can be maintained using the makeup strategies described in Phase 2. However, NSRC equipment is available during Phase 3 for SFP cooling to provide additional defense-in-depth and a backup to the on-site FLEX Pumps.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

Condition 6 of NEI 12-06, Section 3.2.1.3, states that permanent plant equipment contained in structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles, are available. In addition, Section 3.2.1.6 states that the initial SFP conditions are: 1) all boundaries of the SFP are intact, including the liner, gates, transfer canals, etc., 2) although sloshing may occur during a seismic event, the initial loss of SFP inventory does not preclude access to the refueling deck around the pool and 3) the SFP cooling system is intact, including attached piping.

The FIP indicates that boiling begins at approximately 10 hours during a full-core offload, outage situation, and the staff finds this scenario bounds the normal operation, non-outage situation. The staff noted that the licensee's sequence of events timeline in its FIP indicates that operators will deploy hoses and spray nozzles as a contingency for SFP makeup within 6 hours after the initiating event to ensure the SFP area remains habitable for personnel entry.

As described in its FIP, the licensee's Phase 1 SFP cooling strategy does not require any operator actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of the SFP. The operators are directed to open specified doors in the Reactor Building to passively cooldown the different elevations in the Reactor Building and channel potentially contaminated air/steam mixture from the Reactor Building roof to atmosphere.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves the use of the FLEX Pump or NSRC supplied pump for Phase 3, with suction from the intake/discharge canal water from Barnegat Bay, to supply water to the SFP. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX pump is discussed in SE Section 3.7.3.1 below. Furthermore, the staff's evaluation of the robustness and availability of the intake/discharge canal water from Barnegat Bay for an ELAP event is discussed in Section 3.10.

3.3.4.1.2 Plant Instrumentation

In its FIP, the licensee stated that the instrumentation for SFP level will meet the requirements of Order EA-12-051. Furthermore, the licensee stated that these instruments will have initial local battery power with the capability to be powered from the FLEX DGs. The NRC staff's review of the SFP level instrumentation, including the primary and back-up channels, the display to monitor the SFP water level and environmental qualifications to operate reliably for an extended period are discussed in Section 4 of this SE.

3.3.4.2 Thermal-Hydraulic Analyses

In the FIP, Section 2.5.6 states the normal SFP water level at the start of the event is approximately 23 ft. over the top of the spent fuel seated in the storage racks. The staff noted the licensee used its design-basis maximum heat load, as documented in UFSAR Section 9.1.3.2.3, that indicates that the SFP water inventory will heat up from 125°F to 212°F during the first 10.3 hours with a boil-off of the water inventory equivalent to 41.2 gpm. Starting at 23 ft. above the fuel, it will take approximately 35.25 hours to reach a level 10 ft. above the spent fuel seated in the storage racks.

Based on the licensee's sequence of events in FIP Section 2.17, the staff noted that 6 hours after the initiating event, personnel will be procedurally directed to begin staging equipment for fuel pool makeup and spray prior to the SFP reaching the boiling point. Furthermore, 12 hours after the initiating event, the ability to provide makeup/spray to the SFP using the FLEX Pump is expected to begin. The licensee explained that based on a full-core offload, less than a foot (ft.) of water would be boiled-off by the time the FLEX Pump can provide water to the SFP. Thus, based on the expected SFP boiling time, the staff finds the licensee has sufficient time to deploy, stage and begin providing water to the SFP.

The staff finds that the licensee conservatively determined that a SFP makeup flow rate of at least 41.2 gpm will maintain adequate SFP level above the TAF for an ELAP occurring during normal power operation. In NEI 12-06, Section 3.2.1.6 states that the SFP heat load assumes the maximum design-basis heat load for the site as one of the initial SFP conditions. Consistent with this guidance in NEI 12-06, Section 3.2.1.6, the staff finds the licensee has considered the maximum design-basis SFP heat load.

3.3.4.3 FLEX Pumps and Water Supplies

In the FIP, Section 2.5.7.1 states that 41.2 gpm is required for SFP makeup to account for the calculated boiling rate of the pool for a full core offload (UFSAR Section 9.1.3.2.3). Per the licensee's hydraulic calculation, the staff noted that the FLEX Pump can deliver a minimum makeup rate of 269 gpm (at 52 psig) to the SFP, which is inclusive of the 41.2 gpm normal boil off rate. The licensee explained that its hydraulic calculation demonstrates for the bounding scenario that the FLEX Pump with suction from the intake/discharge canal water from Barnegat Bay will simultaneously supply makeup to all three users (i.e., flow path to the IC, RPV and SFP) providing a margin of 227.8 gpm. In addition, during its audit, the staff reviewed the licensee's hydraulic calculation and confirmed that when determining the available makeup rate of 269 gpm to the SFP, the licensee accounted for the characteristics for the spray nozzles; thus, the licensee's calculation determined that an excess of 250 gpm of spray flow can be delivered for SFP cooling.

As described in the FIP, the SFP cooling strategy relies on the FLEX Pump to provide SFP makeup during Phase 2. In the FIP, Section 2.5.7.1 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX Pump. Specifically the pump is trailer mounted Dri-Prime HL 130M Godwin Pump rated at approximately 1100 gpm at 150 psid and is a self-priming, variable speed, diesel engine driven centrifugal pump. The staff noted that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3, as described in FIP Table 5, would allow the NSRC pump to fulfill the mission of the onsite FLEX pump if the onsite

FLEX pump were to fail. As stated above, the FLEX Pump can provide a SFP makeup rate of 269 gpm and SFP spray rate in excess of 250 gpm, which both meet or exceed the maximum SFP makeup requirements. Furthermore, the staff finds analysis above is consistent with NEI 12-06 Section 11.2 and the FLEX equipment is capable of supporting the SFP cooling strategy and is expected to be available during an ELAP event.

3.3.4.4 Electrical Analyses

The licensee's Phase 1 electrical strategy is to monitor SFP level using instrumentation installed as required by NRC Order EA-12-051 (the capability of this instrumentation is described in Section 4 of this SE). In its FIP, the licensee stated that the SFP level instrumentation has a battery backup that will provide power to the instrumentation for greater than 72 hours when normal power is lost.

The licensee's Phase 2 electrical strategy is to continue monitoring SFP level and repower SFP level instrumentation using the 480 Vac FLEX DG.

The licensee's Phase 3 electrical strategy is to continue with the Phase 2 strategy and substitute the 480 Vac NSRC-supplied CTG for the Phase 2 FLEX DG, if necessary.

The NRC staff reviewed licensee A/R 2386806 and determined that the 480 Vac FLEX DGs should have sufficient capacity and capability to supply SFP level instrumentation. The NRC staff also finds that the 480 Vac CTGs being supplied by an NSRC have adequate capacity and capability since they are of higher capacity than the FLEX DGs (1100 kW versus 500 kW).

Based on its review, the NRC staff finds that the licensee's strategy is acceptable to power and monitor SFP level instrumentation during an ELAP as a result of a BDBEE.

3.3.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately should maintain or restore SFP cooling following an ELAP consistent with NEI 12-06 guidance as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.4 Containment Function Strategies

The industry guidance document, NEI 12-06, Table 3-1, provides examples of acceptable approaches for demonstrating the baseline capability of the containment strategies to effectively maintain containment functions during all phases of an ELAP event. One such approach is for a licensee to perform an analysis demonstrating that containment pressure control is not challenged. Oyster Creek has a General Electric Boiling Water Reactor with a Mark I containment.

The licensee performed a containment evaluation, OC-MISC-010, "MAAP Analysis to Support FLEX Initial Strategy," Revision 1, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy for removal of decay heat from the RPV utilizing the IC as well as isolating containment and monitoring containment pressure and temperature and concluded that the containment parameters of pressure and temperature remain well below the respective UFSAR Section 6.2.1.1 design limits of 35 psig and 281 °F for the drywell and 35 psig and 150 °F for the suppression pool (Torus) for more than 72 hours. From its review of the evaluation, the NRC staff noted that the required actions to maintain containment integrity and required instrumentation functions have been developed, and are summarized below.

3.4.1 Phase 1

The FIP states during Phase 1, primary containment integrity is maintained by normal design features of the containment, such as the containment isolation valves. In accordance with NEI 12-06, the containment is assumed to be isolated following the event. Two of five EMRVs automatically cycle to control reactor pressure until the ICs are automatically placed into service. Station loss of offsite power (LOOP)/SBO procedure ABN-36 will be used to support immediate actions in FLEX core cooling Phase 1. The ICs automatically initiate, and operators' achieve a ≤10°F/hr cooldown rate. Once the potential ELAP condition is recognized (≤10 minutes into the event), RPV cooldown rate will increase to 50°F/hour. The ICs will remove the decay heat energy from the RPV and return the water to the RPV, therefore, its use conserves vessel inventory. The energy deposited to the containment is from radiative heat transfer, leakage from the reactor recirculation pump seals, and unidentified reactor system leakage to the containment. As a result, containment parameters are not challenged and venting is not anticipated.

3.4.2 Phase 2

During Phase 2, two IC loops continue to operate so that decay heat is removed from the RPV thus limiting the heat released to the primary containment. Transition for FLEX Phase 1 to FLEX Phase 2 is relatively rapid. Once a potential ELAP/LUHS event is recognized (≤10 minutes), the ICs will be manually placed into service, and operators achieve a 50°F/hour cooldown rate. At 90 minutes from ELAP recognition, makeup water to the ICs is available. Primary IC makeup is accomplished by providing a 5" hose from a portable FLEX pump to a connection point on the reactor building 23 -ft FLEX manifold. From this FLEX manifold, flow will go through a mechanical riser to a second FLEX manifold on reactor building 95-ft. elevation. From the reactor building 95-ft. elevation FLEX manifold, makeup flow will travel via a 3-in. hose to a common drain connection of the ICs, valve V-14-132. The alternate make up method is from the reactor building 23-ft. FLEX manifold to valve V-11-63. Based on evaluation OC-MISC-010, the licensee expects the containment temperature and pressure to remain below limits.

3.4.3 Phase 3

The FIP states that the Phase 3 strategy to reduce containment temperature and pressure and to ensure continued functionality of the key parameters will utilize existing plant systems restored by off-site equipment and resources. During Phase 3, the ICs will remain in service with IC make-up continuing to be provided by Phase 2 portable equipment and backed up by NSRC pumps. The containment pressure will be monitored and, if necessary, the primary containment will be vented using existing procedures and systems. Based on evaluation OC-MISC-010, the licensee expects the containment temperature and pressure to remain below limits and venting the primary containment is not required.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

Guidance document NEI 12-06 baseline assumptions have been established on the presumption that other than the loss of the ac power sources and normal access to the UHS, installed equipment that is designed to be robust with respect to design-basis external events is assumed to be fully available. Installed equipment that is not robust is assumed to be unavailable. Below are the baseline assumptions for the availability of SSCs for maintaining containment functions during an ELAP.

3.4.4.1.1 Plant SSCs

Containment

Section 3.8.2.1 of the Oyster Creek UFSAR, Revision 16, describes a General Electric Mark I pressure suppression system with two large chambers. The drywell houses the reactor vessel, the reactor coolant recirculating loops, and other components associated with the reactor system. The UFSAR Section 6.2.1.1 states that the internal design pressure is 35 psig and the internal maximum design temperature is 281 °F.

The pressure absorption chamber (wetwell or torus) is a steel shell in the shape of a torus located below and around the base of the drywell. The UFSAR Section 6.2.1.1 states that the internal design pressure is 35 psig and the internal maximum airspace design temperature is 150 °F.

The two chambers are interconnected through vent pipes which feed into a common header withdowncomer pipes extending below the water level in the pressure absorption chamber. Gas phase return lines with vacuum breaker valves feed back gas to the drywell in case its pressure is less than the absorption chamber.

The containment is designed as a seismic Class I structure. The staff noted that being a seismic Category I structure, the containment is reasonably protected from all beyond-design-basis external hazards and expected to maintain its function.

Isolation Condenser

The primary function of the IC system is to remove reactor decay heat when the vessel is isolated from the main heat sink. Following vessel isolation, the reactor pressure will rise causing EMRVs to cycle to control reactor pressure below safety valve set points. Normally, both IC loops are placed into operation automatically by opening the condensate return valves. In an SBO condition the operators are directed to confirm closure of the MSIVs, and manually control reactor pressure using both ICs. This will minimize loss of inventory through the EMRVs and downstream steam loads. Also, this will result in lowering reactor pressure to minimize coolant loss from leakage and energy addition into primary containment.

The IC system operates by natural circulation without the need for driving power other than the dc electrical system used to place the IC system in operation. The system operates with steam flowing from the RPV through the condenser tubes and condensate returning by gravity to the

RPV, forming a closed loop. The valves in the steam inlet lines are normally open so that the tube bundles are at reactor pressure. Only the dc motor operated condensate isolation valves are normally closed.

The design heat removal capacity of the IC system (two condensers) is 410 x 10^6 Btu/hr. The IC system design provides for 100 minutes of continuous operation with both ICs available and without shell-side makeup. After 100 minutes, make up to the ICs shells is necessary to keep the shell-side level above the tube bundles. For the FLEX strategy, makeup to the shell-side of the ICs will be provided within 90 minutes of commencing ELAP preparations which occurs at ≤ 10 minutes. This is accomplished by placing into service a FLEX pump taking suction from the Intake/Discharge Canal (UHS) that will be capable of providing make up to the ICs shells allowing for indefinite operation.

3.4.4.1.2 Plant Instrumentation

In NEI 12-06, Table 3-1, specifies that containment pressure, suppression pool level, and suppression pool temperature are key containment parameters which should be monitored by repowering the appropriate instruments.

The FIP identifies the following as key parameters credited for all phases of the strategy for maintaining containment integrity:

- Torus Water Temperature: TI-664-42A, TI-664-428
- Torus Narrow Range Water Level: IP10B
- Drywell Pressure Narrow Range: PI-642-009A, PI-642-0098
- RPV Level: FZ level - C & D: LI-622-1001. LI-622-1002
- RPV Wide Range Pressure -C & D: LI-622-1018. LI-622-1019
- IC Shell Level: A & B: LT-IG0006A and LT-IG0006B

The above instrumentation is available prior to and after dc load shedding of the dc buses during SBO/ELAP response procedure implementation. In the unlikely event that the battery bus infrastructures or supporting equipment is damaged and non-functional rendering key parameter instrumentation unavailable in the Oyster Creek main control room, FLEX strategy guidelines for alternately obtaining the critical parameters locally is provided in FSG-20, "EOP Key Parameter-Alternate Instrumentation Reading".

Based on this information, the licensee should have the ability to appropriately monitor the key containment parameters as delineated in NEI 12-06, Table 3-1.

3.4.4.2 Thermal-Hydraulic Analyses

The licensee performed a containment evaluation, OC-MISC-010, "MAAP Analysis to Support FLEX Initial Strategy," Revision 1, which was based on the boundary conditions described in Section 2 of NEI 12-06. This calculation utilized the MAAP computer code, version 4.0.5, to perform numeric computations of the fundamental thermodynamic equations which predict the heat up and pressurization of the containment atmosphere under ELAP conditions.

The calculation shows the drywell temperature reaches 261 °F and the suppression pool temperature reaches 98 °F at 72 hours. The drywell pressure reaches a maximum of 15.1 psig and the suppression pool pressure reaches a maximum of 13.5 psig during the 72-hour period. The maximum values calculated are well below the UFSAR design parameters stated above in Section 3.4, so the licensee has adequately demonstrated that there is significant margin before a limit would be reached.

3.4.4.3 FLEX Pumps and Water Supplies

In the FIP, Section 2.4 explains that during Phase 1, normal design features of the containment maintain primary containment integrity and the ICs remove the decay heat from the RPV and return the water to the RPV. During Phase 2, the two IC loops continue to operate to remove decay heat with makeup water being provided by the FLEX Pump; thus, limiting the heat released to the primary containment.

The staff finds it reasonable that no mitigation actions are necessary to maintain or restore containment cooling during Phases 1 or 2 based on the licensee's actions to provide makeup to the ICs and RPV and the expected response of the containment during an ELAP event. In addition, the staff noted that the licensee's containment integrity strategies do not rely on the use of FLEX pumps and water sources for maintaining containment pressure or temperature below the design limits for at least 72 hours allowing for the arrival of off-site resources to assist in restoring containment cooling. Phase 3 equipment can be used to support containment cooling if necessary.

3.4.4.4 Electrical Analyses

The licensee performed a containment evaluation based on the boundary conditions described in Section 2 of NEI 12-06. Based on the results of its evaluation, the licensee developed required actions to ensure maintenance of containment integrity and required instrumentation continues to function. With no cooling in the containment, temperatures in the containment are expected to rise. The licensee's Phase 1 coping strategy includes monitoring containment temperature and pressure using installed instrumentation powered by the Class 1E station batteries. The licensee's strategy to repower instrumentation using the Class 1E station batteries is described in Section 3.2.3.6 of this SE and is adequate to ensure continued containment monitoring.

The licensee's Phase 2 coping strategy is to continue the Phase 1 coping strategy (i.e monitor containment temperature and pressure using installed instrumentation). The licensee's strategy to repower instrumentation using the 480 Vac, 500 kW FLEX DGs is described in Section 3.2.3.6 of this SE. This DG is adequately sized to ensure continued containment monitoring.

The licensee's Phase 3 coping strategy is to continue the Phase 2 strategy with equipment supplied by an NSRC as a backup. If necessary, an NSRC supplied 480 Vac CTG could replace the Phase 2 480 Vac FLEX DG. The Phase 3 equipment can also be used to initiate containment cooling.

Based on its review, the NRC staff determined that the electrical equipment available onsite (e.g., 480 Vac FLEX DGs) supplemented with the equipment that will be supplied from an NSRC (e.g., 480 Vac CTG), provides sufficient capacity and capability to supply power to support the licensee's strategies.

3.4.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain or restore containment functions following an ELAP event consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.5 Characterization of External Hazards

Sections 4 through 9 of NEI 12-06 provide the methodology to identify and characterize the applicable BDBEEs for each site. In addition, NEI 12-06 provides a process to identify potential complicating factors for the protection and deployment of equipment needed for mitigation of applicable site-specific external hazards leading to an ELAP and loss of normal access to the UHS.

Characterization of the applicable hazards for a specific site includes the identification of realistic timelines for the hazard, characterization of the functional threats due to the hazard, development of a strategy for responding to events with warning, and development of a strategy for responding to events without warning.

The licensee reviewed the plant site against NEI 12-06 and determined that FLEX equipment should be protected from the following hazards: seismic; external flooding; severe storms with high winds; snow, ice and extreme cold; and extreme high temperatures.

References to external hazards within the licensee's mitigating strategies and this SE are consistent with the guidance in NEI-12-06 and the related NRC endorsement of NEI 12-06 in JLD-ISG-2012-01. Guidance document NEI 12-06 directed licensees to proceed with evaluating external hazards based on currently available information. For most licensees, this meant that the OIP used the current design basis information for hazard evaluation. Coincident with the issuance of Order EA-12-049, on March 12, 2012, the NRC staff issued a Request for Information pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) [Reference 19] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies. Due to the time needed to reevaluate the hazards, and for the NRC to review and approve them, the reevaluated hazards were generally not available until after the mitigation strategies had been developed. The NRC staff has developed a proposed rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was published for comment in the *Federal Register* on November 13, 2015 [Reference 43]. The proposed MBDBE rule would make the intent of Orders EA-12-049 and EA-12-051 generically applicable to all present and future power reactor licensees, while also requiring that licensees consider the reevaluated hazard information developed in response to the 50.54(f) letter.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in response to the 50.54(f) letter and the requirements for Order EA-12-049 and the MBDBE rulemaking (see COMSECY-14-0037, Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards" [Reference 42]). The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 20]. The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 [Reference 37], the NRC staff informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the NRC SEs and inspections related to Order EA-12-049 will rely on the guidance provided in JLD-ISG-2012-01, Revision 0, and the related industry guidance in NEI 12-06, Revision 0. The hazard reevaluations may also identify issues to be entered into the licensee's corrective action program consistent with the OIPs submitted in accordance with Order EA-12-049.

As discussed above, licensees are reevaluating the site seismic and flood hazards as requested in the NRC's 50.54(f) letter. After the NRC staff approves the reevaluated hazards, licensees will use this information to perform flood and seismic mitigating strategies assessments (MSAs) per the guidance in NEI 12-06, Revision 2, Appendices G and H [Reference 6]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 7]. The licensee's MSAs will evaluate the mitigating strategies described in this SE using the revised seismic hazard information and, if necessary, make changes to the strategies or equipment. Licensees will submit the MSAs for NRC staff review.

The licensee developed its OIP for mitigation strategies by considering the guidance in NEI 12-06 and the site's design-basis hazards. Therefore, this SE makes a determination based on the licensee's OIP and FIP. The characterization of the applicable external hazards for the plant site is discussed below.

3.5.1 Seismic

In its FIP, the licensee described the current design-basis seismic hazard, the SSE. Per the Oyster Creek UFSAR, the original SSE for seismic design was based on a Housner-shaped ground response spectrum and peak ground acceleration (PGA) of 0.22g derived from the El Centro earthquake ground acceleration scaled to account for the regional seismic activity appropriate for New Jersey. The licensee stated that the UFSAR was updated to specify that starting in September 1995, the Weston Geophysical Site Specific Response spectrum (SSAS) and associated in-Structure Response Spectra (ISRS) would be used as the design and licensing bases SSE going forward. The 1995 Weston SSAS anchored at 0.184g horizontal and 0.0952g vertical (vertical acceleration as stated in the Integrated Plan) PGA represents the lower bound capacity earthquake for plant safe shutdown and is therefore appropriate for use in the seismic screening. It should be noted that the actual seismic hazard involves a spectral graph of the acceleration versus the frequency of the motion. Peak acceleration in a certain frequency range, such as the numbers above, is often used as a shortened way to describe the hazard

As the licensee's seismic reevaluation activities are completed, the licensee is expected to assess the mitigation strategies to ensure they can be implemented under the reevaluated hazard conditions as will potentially be required by the proposed MBDBE rulemaking. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.2 Flooding

In its FIP, the licensee described nine potential flood causing mechanisms that were evaluated for OCNGS. The evaluation showed two bounding events for the limiting site flooding levels. The current licensing basis maximum flood level due to the probable maximum hurricane (PMH) is elevation 22-ft. mean sea level (MSL) with waves at plant site of up to 1 ft. high. The plant grade, elevation 23-ft. MSL, is one ft. above the still water PMH flood level. The licensee stated that the flood will not find its way into the plant buildings, the floor levels of which are generally 6-inches (in.) above grade at elevation 23-ft. 6-in. The intake structure with its deck at elevation 6-ft. MSL will be under water. This deck supports, apart from the other equipment, the circulating water pumps, service water pumps and the emergency service water pumps. During a PMH flood, the circulating water, service water pumps and the emergency service water pumps will become inoperable and thus emergency plant procedures have been instituted which require the plant to be shutdown when flood waters reach a predetermined level as to ensure the capability for safe shutdown under either normal or abnormal conditions.

The other flood causing mechanism is the probable maximum precipitation (PMP) event. The flooding level which would be caused by local PMP is 23.5-ft. MSL. The two entrances to the EDG building are at elevation 23-ft. MSL, which is 6 in. below the PMP. A 6 in. high asphalt dike is provided at these entrances to provide protection against internal flooding of the EDG building.

In its FIP, the licensee concluded that the current design-basis is that the Seismic Category I structures are not susceptible to external flooding from the PMH or PMP events. During the on-site audit the licensee noted that per UFSAR Section 3.8.5.1 the exterior surfaces of walls and slabs have waterproofing materials applied to seal against intrusion of groundwater. During the on-site audit the licensee further stated that there are no non-seismically robust large internal flooding sources in the reactor building that would impede the mitigating strategies.

As the licensee's flooding reevaluation activities are completed, the licensee is expected to assess the mitigation strategies to ensure they can be implemented under the reevaluated hazard conditions as will potentially be required by the proposed MBDBE rulemaking. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.3 High Winds

In NEI 12-06, Section 7, provides the NRC-endorsed screening process for evaluation of high wind hazards. This screening process considers the hazard due to hurricanes and tornadoes.

The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06, Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009); if the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 mph exceeds 1E-6 per year, the site should address hazards due to extreme high winds associated with hurricanes using the current licensing basis for hurricanes.

The screening for high wind hazard associated with tornadoes should be accomplished by comparing the site location to NEI 12-06, Figure 7-2, from U.S. NRC, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461, Revision 2, February 2007; if the recommended tornado design wind speed for a 1E-6/year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes using the current licensing basis for tornados or Regulatory Guide 1.76, Revision 1.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 39° 49' North latitude and 74° 12' West longitude. In NEI 12-06 Figure 7-1, "Contours of Peak-Gust Wind Speeds at 10-m Height in Flat Open Terrain, Annual Exceedance Probability of 10^{-6}," indicates the site is in a region where the hurricane wind speed exceeds 130 mph. In NEI 12-06 Figure 7-2, "Recommended Tornado Design Wind Speeds for the 1E-6/year Probability Level" indicates the site is in a region where the tornado design wind speed exceeds 130 mph. Therefore, the plant screens in for an assessment for high winds caused by hurricanes and tornados, including missiles produced by these events.

Therefore, high-wind hazards are applicable to the plant site. The licensee has appropriately screened in the high wind hazard and characterized the hazard in terms of wind velocities and wind-borne missiles.

3.5.4 Snow, Ice, and Extreme Cold

As discussed in NEI 12-06, Section 8.2.1, all sites should consider the temperature ranges and weather conditions for their site in storing and deploying FLEX equipment consistent with normal design practices. All sites outside of Southern California, Arizona, the Gulf Coast and Florida are expected to address deployment for conditions of snow, ice, and extreme cold. All sites located north of the 35th Parallel should provide the capability to address extreme snowfall with snow removal equipment. Finally, all sites except for those within Level 1 and 2 of the maximum ice storm severity map contained in Figure 8-2 should address the impact of ice storms.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at N 39° 49' and W 74° 12'. In addition, the site is located within the region characterized by EPRI as ice severity level 3 (NEI 12-06, Figure 8-2, Maximum Ice Storm Severity Maps). Consequently, the site is subject to icing conditions that could impact the mitigating strategies. The licensee concludes that the plant screens in for an assessment for snow, ice, and extreme cold hazard.

In summary, based on the available local data and Figures 8-1 and 8-2 of NEI 12-06, the plant site does experience significant amounts of snow, ice, and extreme cold temperatures; therefore, the hazard is screened in. The licensee has appropriately screened in the hazard and characterized the hazard in terms of expected temperatures.

3.5.5 Extreme Heat

In the section of its FIP regarding the determination of applicable extreme external hazards, the licensee stated that, as per NEI 12-06 Section 9.2, all sites are required to consider the impact of extreme high temperatures. Per UFSAR Section 2.3 Meteorology Table 2.3.2, the maximum recorded temperature at Oyster Creek as recorded at Pleasantville NJ was 106°F July of 1936. The plant site screens in for an assessment for extreme high temperature hazard.

In summary, based on the available local data and the guidance in Section 9 of NEI 12-06, the plant site does experience extreme high temperatures. The licensee has appropriately screened in the high temperature hazard and characterized the hazard in terms of expected temperatures.

3.5.6 Conclusions

Based on the evaluation above, the NRC staff concludes that the licensee has developed a characterization of external hazards that is consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order in regard to the characterization of external hazards.

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

In its FIP, the licensee described the onsite FLEX storage areas as consisting of two outdoor concrete storage pads. The FLEX storage pads are located inside the protected area at the far northwest and southeast locations of the plant. Stored on these pads are the two FLEX diesel driven pumps, two diesel driven generators, two hose trailers, as well as sea van containers containing other miscellaneous supporting equipment such as fuel transfer pumps and hoses, small portable diesel driven generators, miscellaneous hand tools and debris removal tools. Additional hoses and cables are stored inside the turbine building and the reactor building. The debris removal and tow vehicles are stored on these concrete pads. The towing and debris removal equipment consists of one F750 truck and one Kubota tractor. The licensee stated that the storage pad locations were chosen to keep the FLEX equipment from being blocked from downed power lines since downed power lines are one of the most likely site damage scenarios that can delay FLEX deployment.

Below are additional details on how FLEX equipment is protected from each of the applicable external hazards.

3.6.1.1 Seismic

In its FIP, the licensee stated that the FLEX storage pads are designated as seismic Category I. The outdoor FLEX storage pads are constructed and evaluated to withstand the maximum potential earthquake stresses for the site. Large FLEX portable equipment such as the diesel driven pumps and generators, tow and debris removal vehicles, and hose trailers, are secured with tie-down straps to seismic block anchors to protect them during a seismic event and to prevent seismic interaction with nearby components. The northwest and southeast pads are each located as to avoid seismic interaction with buildings in the area. All equipment credited for implementation of the FLEX strategies is either stored at the FLEX storage pads or in a plant structure that meets the station's design bases for SSE, specifically the reactor building and turbine building, where additional hoses and cables are stored.

3.6.1.2 Flooding

The two most limiting events for flooding are a hurricane storm surge with driven wave run up and the probable maximum precipitation event. Upon receiving a hurricane warning the licensee plans to relocate the FLEX equipment to indoor locations. The licensee stated that flooding due to the PMH is therefore bounded by high wind, hurricane conditions, which are discussed below.

In its FIP, the licensee stated that the PMP event exceeded the height of the entrances to the EDG building. Consequently, the licensee constructed a 6 in. high asphalt dike at these entrances to provide protection against internal flooding of the EDG building. Although no FLEX equipment is stored in the EDG building, access is required for refueling of the FLEX pumps and generators.

3.6.1.3 High Winds

Large FLEX equipment pumps and generators (N and N+1) are normally stored on the outdoor FLEX storage pads. The licensee stated that during predicted hurricane conditions, site procedures will relocate the large FLEX equipment to site truck bays. The N set of large FLEX equipment will be moved into the turbine building truck bay, a robust building meeting the plant's design-basis for high wind hazards. The N set of cable and hoses that are pre-staged in the turbine and reactor buildings will not need to be relocated. In its FIP, the licensee stated that the turbine and reactor buildings are robust buildings meeting the plant's design-basis for high wind hazards. The N+1 set of large FLEX equipment will be relocated to the low level radwaste (LLRW) truck bay.

In its FIP, the licensee stated that the FLEX storage pads are located in such a manner as to avoid damage to both sets of FLEX equipment in the case of a tornado. The pads are located approximately 1200 ft. apart. The locations for the pads considered the typical tornado path for Ocean County. The north to south orientation of the two storage paths provides a separation distance of approximately 1050 ft. when the site is approached along the typical tornado path. The licensee stated that based on historical data the maximum recorded tornado width in Ocean County was 750 ft. and the typical approach path is from the west southwest.

FLEX equipment stored at the pads is secured to prevent damage or becoming airborne due to the tornado winds.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee stated that FLEX equipment is protected from severe temperatures. As described in the FIP, each storage pad will have electrical connections available to provide power to FLEX equipment battery trickle chargers and block heaters. Power is provided to the two pads from different sources. Therefore, should one electrical source fail the other electrical source will be available. For predicted severe snow and ice weather events, plant procedures will direct the relocation of the N set of FLEX equipment to the turbine building truck bay. This location meets the plant's design-basis for the snow, ice and cold conditions. The licensee stated that diesel fuel for FLEX equipment is treated with a fuel additive during cold weather conditions to prevent gelling.

Portable diesel equipment and portable static battery chargers will be stored and operated in areas outdoors. Licensee reviewed the vendor's information to verify that the equipment is expected to operate at the high ambient temperatures.

3.6.1.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should protect the FLEX equipment during a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.6.2 Availability of FLEX Equipment

Section 3.2.2.16 of NEI 12-06 states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare (i.e., an N+1 capability, where "N" is the number of units on site). It is also acceptable to have a single resource that is sized to support the required functions for multiple units at a site (e.g., a single pump capable of all water supply functions for a dual unit site). In this case, the N+1 could simply involve a second pump of equivalent capability. In addition, it is also acceptable to have multiple strategies to accomplish a function, in which case the equipment associated with each strategy does not require an additional spare.

Since Oyster Creek is a single unit site, two sets of FLEX equipment are provided. In the FIP, Table 4 lists two FLEX diesel driven pumps, two FLEX diesel driven generators, two fuel oil (FO) transfer rigs, six Yanmar portable diesel driven generators, two hose trailers and two tow and debris removal vehicles. The hose trailers contain the N and the N+1 hoses required to connect a FLEX pump to the UHS and connect the pump discharge to the manifold located on the 23 ft. level in the reactor building. Additionally, the hoses used to connect the two manifolds (one on the 23 ft. elevation and one on the 95 ft. elevation) to various system tie in points are stored in the turbine and reactor building. The primary N set of six cables with color coded connectors is stored in the turbine building north mezzanine on reels inside designated cabinets. The alternate N+1 set of cables is stored in the two sea vans (three per sea van) at the storage pads. Per the FIP, each cable is 350-foot in length. Therefore, storing 3, N+1 cables at each storage pad provides an additional 50 percent of cabling at each pad which exceeds the additional 10 percent of cabling as described in NEI 12-06, Section 3.2.2.16.

Based on the number of portable FLEX pumps, FLEX DGs, and support equipment identified in the FIP and during the audit review, the NRC staff finds that, if implemented appropriately, the licensee's FLEX strategies include a sufficient number of portable FLEX pumps, FLEX DGs, and equipment for RPV makeup and core cooling, SFP makeup, and maintaining containment consistent with the N+1 recommendation in Section 3.2.2.16 of NEI 12-06.

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee has identified both primary and alternate deployment paths from each storage pad to the discharge canal for staging the FLEX diesel driven pump and to the northwest side of the turbine building for staging the FLEX diesel driven generator.

3.7.1 Means of Deployment

The deployment of onsite FLEX equipment to implement coping strategies beyond the initial plant capabilities requires that pathways between the FLEX storage pads and various deployment locations be clear of debris resulting from seismic, high wind (hurricane and tornado), excessive snow/ice, or flooding events. In its FIP, the licensee stated that signs requiring paths and areas to remain clear of obstructions have been posted along deployment pathways and at FLEX equipment deployment locations.

The towing and debris removal equipment includes a F750 truck equipped with a snow removal blade and a Kubota frontend loader. Towing and debris removal equipment is either stored at the FLEX storage pads or during severe weather is relocated into buildings such that the equipment remains functional and deployable to clear obstructions from the pathway between the FLEX equipment storage locations and the deployed staging areas. Debris removal hand tools such as tow chains, diesel-powered chainsaws, axes, sledgehammers, shovels, Cant hook, and bolt cutters are also available. This miscellaneous equipment is stored in the two sea van containers located at each FLEX storage pad.

Snow and ice storms can provide enough buildup to affect travel within the site. However, the licensee stated that reasonable warning time should provide enough time for progressive snow and ice removal by normal means. Clearing of FLEX deployment pathways has been incorporated into the OCNGS snow removal plan. During an event the FLEX frontend loader can be used for snow removal from FLEX deployment pathways along with the F750, which is equipped with a snow removal blade.

Deployments of the FLEX and debris removal equipment from the FLEX storage pads are not dependent on off-site power. Because the FLEX storage pads are outdoors, all actions required to access and deploy debris removal equipment and BDBEE/FLEX equipment can be accomplished manually. The licensee stated in its FIP that the roll up door for the LLRW truck bay has a manual override and does not need electrical power to be opened. All actions are accomplished manually. During the on-site audit the licensee stated that the turbine building rollup door can also be manually opened without reliance on electric power.

3.7.2 Deployment Strategies

The licensee performed an evaluation based on the original soil borings and the soil borings performed for the installation of the security walls around the EDG building and concluded that the consequence of soil liquefaction would not adversely affect the FLEX strategies at Oyster Creek. The licensee has performed additional evaluation with new bore holes to assess the impact of soil liquefaction at the off-site SAFER staging area. The licensee concluded that the consequence of soil liquefaction at the SAFER staging area would not adversely affect the FLEX strategies.

The UHS, Barnegat Bay, is accessed from the intake structure using the FLEX diesel driven pump. The diesel driven FLEX pump will be transported from the storage pad (or from its relocated position in the turbine building or the LLRW building) to either the primary staging area, the chemistry sample point in the discharge canal or the secondary staging area, the intake canal between the traveling screens and the circulating water pumps. A non-collapsible hose and strainer will be routed from the pump suction and lowered into the discharge tunnel. The intake screen and the strainer limit the solid debris size for pump protection. Flexible hose sections will be connected and routed from the FLEX pump discharge to a manifold located in the reactor building.

The licensee stated that ice buildup would not impede access to the UHS since during normal operation the intake canal is heated by recirculating the warm water from the discharge canal. Following an ELAP event during extreme cold weather, the FLEX pump is deployed within 100 minutes thus minimizing the potential for frazil ice to form in the intake or discharge canal.

Guideline FSG-05 gives instructions to stage the pump at a higher elevation than the intake structure during the maximum flooding event. The primary staging area for the FLEX pump is at the intake structure deck, which is 6 ft. above MSL. The intake structure deck will be submerged during the flood event necessitating that the pump be moved to higher ground as the water rises.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

The licensee's strategy to provide water to the RPV, ICs, and SFP involve the use of a FLEX manifold installed on the Reactor Building elevation 23-ft elevation North that will be pressurized by the FLEX Pump. After receiving water from the discharge of the FLEX Pump, the Reactor Building 23-ft elevation FLEX manifold will supply another FLEX manifold on the Reactor Building 95-ft elevation via an installed mechanical riser. The staff identified the licensee's use of the FLEX manifolds as an alternative from the guidance in NEI 12-06, Revision 2, Section 3.2.2, to have primary and alternate connections for the FLEX strategies; specifically, the licensee's ability to supply water to the RPV, ICs, and SFP is entirely dependent on the FLEX manifolds.

During its audit, the licensee justified its approach of using the FLEX manifolds by confirming that both FLEX manifolds, including the associated piping and supports, were designed to Seismic Category 1 criteria (See ECR 14-00025, Revision 3 and Calculation C-1302-917-E310-001, Revision 1). In addition, the licensee stated that it performed walkdowns around the area of the FLEX manifolds and associated riser piping to identify adjacent components that may potentially have adverse seismic interaction following an earthquake. During its walkdown, the licensee identified six instances of possible seismic interaction on varying elevations of the Reactor Building that may adversely impact the FLEX manifolds and associated riser piping. The licensee performed an evaluation for each instance and determined that the adjacent components would not adversely impact the FLEX manifolds and associated riser piping based either on the design, mounting or size of the components (See EC Request 426620). Based on the robust design/installation and location of the FLEX manifolds, and the licensee's evaluation on potential seismic interaction between the manifolds and adjacent components, the staff finds the FLEX manifolds, including the riser piping, is reasonably protected from all applicable external hazards. Therefore, the NRC staff approves this alternative as being an acceptable method of compliance with the order.

The discharge hose from the FLEX pump is routed through the Turbine Building Northwest Access Door across the Turbine Building North Mezzanine to the FLEX manifold in the Reactor Building. During its audit, the licensee explained that the FLEX hose pathway is protected from tornado-generated missiles based on the thickness of the concrete walls surrounding the pathway (See Engineering Change Request 0426629). Furthermore, based on its review of the seismic analysis performed for the Turbine Building, the licensee determined that this FLEX hose deployment pathway is robust and expected to be available following a seismic event.

Reactor Pressure Vessel Discharge Connection Points

In the FIP, Section 2.9.2 states that the primary RPV connection point involves a short run of hard pipe that is installed between the FLEX manifold on the Reactor Building 23-ft elevation and Core Spray System 1. Once the FLEX manifold is pressurized and the necessary valves are manipulated a single operator has the ability to control the flow rate to the primary RPV injection point and provide makeup via a batch feed method.

Furthermore, the FIP states that the alternate RPV connection point involves running two 100-ft. lengths of 3-in. FLEX hose from the FLEX manifold on the Reactor Building 23-ft elevation to the drain line on Core Spray System 2. Once the FLEX hose is connected and appropriate valves are manipulated, a single operator has the ability to control the flow rate to the primary RPV injection point and provide makeup via a batch feed method. In the UFSAR, Table 3.2-1 indicates that the Core Spray System was designed to be Seismic Class I. Furthermore, the primary and alternate connection points, including the FLEX manifold, are located in the Reactor Building, which is Class I structure protected from externally generated missiles.

Given the design and location of the primary and alternate connection points, as described in the above paragraphs, the staff finds that at least one of the connection points should be available to support RPV makeup, via the portable FLEX Pump during an ELAP caused by an external event, consistent with NEI 12-06, Section 3.2.2.

IC Discharge Connection Points

In the FIP, Section 2.9.3 states that the primary IC connection point involves the use of a 100' length of 3" hose routed from the FLEX manifold on the Reactor Building 95-ft elevation to the IC common drain line. Furthermore, as indicated in the FIP and noted by the staff during its audit, the alternate IC connection point involves the use of a 100' length of 3" hose routed from the FLEX manifold on the Reactor Building 23-ft elevation to the tell-tale valve on the makeup to the Emergency Condensers. For either connection point, once the FLEX hose is connected and appropriate valves are manipulated both ICs will receive makeup water from the FLEX Pump. In the UFSAR, Table 3.2-1 indicates that the IC System was designed to be Seismic Class I. Furthermore, the primary and alternate connection points, including FLEX manifold, is located in the Reactor Building, which is a Class I structure protected from externally generated missiles.

Given the design and location of the primary and alternate connection points, as described in the above paragraphs, the staff finds that at least one of the connection points should be available to support IC makeup, via the portable FLEX Pump during an ELAP caused by an external event, consistent with NEI 12-06, Section 3.2.2.

Spent Fuel Pool Discharge Connection Points

In NEI 12-06, Table C-3, states, in part, that the baseline capabilities for SFP Cooling include makeup via hoses on the refuel floor and makeup via connection to SFP cooling piping or other alternate location. Guidance document JLD-ISG-2012-01, Revision 1 states, in part, that spray capability via portable monitor nozzles from the refueling floor using a portable pump is a baseline capability in addition to those identified in NEI 12-06, Table C-3.

In the FIP, Section 2.9.4 states that the primary connection point is established by running a 3" hose from the FLEX manifold on the Reactor Building 95-ft elevation to the SFP diffuser connection located on the Reactor Building 75-ft. elevation. The use of this connection point provides the ability to supply the SFP with makeup water without accessing the refueling floor in the event it is uninhabitable. Furthermore, the FIP states that the alternate connection point involves running 3-in. hose from the FLEX manifold on the Reactor Building 95-ft. elevation to the SFP located on the Reactor Building 119-ft. elevation. During its audit, the staff noted that procedural guidance is provided to the operators to either attach the discharge of the 3-in. hose can to a spray nozzle stored on the refuel floor or securely tie the end of the hose to the SFP railing, which will discharge into the SFP.

Given the design and location of the primary and alternate connection points, as described in the above paragraphs, the staff finds that at least one of the connection points should be available to support SFP makeup/spray, via the portable FLEX Pump during an ELAP caused by an external event, consistent with NEI 12-06, Section 3.2.2. The licensee's FLEX strategy includes the baseline capabilities to provide makeup and spray to the SFP from the refuel floor and makeup to the SFP without accessing the refuel floor consistent with NEI 12-06, Table C-3, and JLD-ISG-2012-01, Revision 1.

3.7.3.2 Electrical Connection Points

Electrical connection points are only applicable for Phases 2 and 3 of the licensee's mitigation strategies for a BDBEE.

During Phase 2, the licensee's strategy is to supply power to equipment required to maintain or restore core cooling, containment, and SFP cooling using a combination of permanently installed and portable components. The licensee's strategy is to use the portable 480 Vac FLEX DGs (N or N+1) stored on the FLEX Storage Pads located at the northwest and southeast quadrants of the site inside the protected area. The 480 Vac FLEX DGs would be deployed to northwest side of the turbine building to provide power to 480 Vac buses USS 1A2 or 1B2. Closing USS tie breaker US2T allows both buses to be powered by the FLEX DG. Procedure FSG-07, "Line-Up of USS 1A2/1B2 for Repowering from FLEX Generator, Revision 1, provide direction to cross-tie 480 Vac buses USS 1A2 and 1B2.

The licensee's primary strategy is to use temporary FLEX cables with color-coded connections staged inside a protected pathway in the turbine building to connect the FLEX DG to a FLEX back feed device located in the "A" 480 Vac room. The FLEX back feed device will be installed in a breaker cubicle located in USS 1A2. The licensee's alternate strategy is to use the same temporary FLEX cables (as the primary strategy) to connect the FLEX DG to a FLEX back feed device located in the "B" 480 Vac room. The FLEX back feed device will be installed in a breaker cubicle located in USS 1B2. Procedure FSG-07, provides guidance for installing the FLEX back feed device in USS 1A2 and 1B2. Procedure FSG-06, "FLEX 480V Cable Deployment and Connection," Revision 0, provides guidance for deploying the FLEX cables and connecting them to the FLEX DG and the FLEX back feed device. The licensee verified proper phase rotation as part of its post installation testing.

For Phase 3, the licensee will receive two 1-MW 4160 Vac CTGs and one 1100 kW 480 Vac CTG from an NSRC. The licensee plans to only connect the 480 Vac CTG and not the 4160 Vac CTGs. The NSRC supplied 480 Vac CTG will be staged in the vicinity of the portable 480 Vac FLEX DG and provide backup power, if necessary. Procedure FSG-23, "Transfer of FLEX Strategy to Safer," provides direction for connecting and verifying proper phase rotation when powering equipment from the NSRC supplied 480 Vac CTG.

3.7.4 Accessibility and Lighting

The FIP described that the doors and gates serve a variety of barrier functions on the site. One primary function is security and is discussed below. However, other barrier functions include fire, flood, radiation, ventilation, tornado, and high energy line break. As barriers, these doors and gates are typically administratively controlled to maintain their function as barriers during normal operations. Following a BDBEE and subsequent ELAP/LUHS event, FLEX coping strategies require the routing of hoses and cables to be run through various barriers in order to connect FLEX equipment to station fluid and electrical systems. For this reason, certain barriers (gates and doors) will be opened and remain open. The licensee stated that this violation of normal administrative controls is acknowledged and is acceptable during the implementation of FLEX coping strategies.

The licensee performed a walkdown to evaluate the adequacy of supplemental lighting and for the practicality of using portable lighting to perform FLEX strategy actions. This walkdown included travel paths to various areas necessary to implement the FLEX strategies, making required mechanical and electrical connections, performing instrumentation monitoring, and component manipulations.

In its FIP, the licensee stated that battery powered Appendix "R" emergency lights were determined to provide adequate lighting for most primary FLEX connection points in the FLEX strategies including the illumination of interior travel pathways needed to access the connection points. These emergency lights are designed and periodically tested to insure the battery pack will provide a minimum of eight hours of lighting with no external ac power sources. The licensee stated that for the FLEX areas requiring lighting not covered by Appendix "R" lights, those areas will have available lighting using non-appendix "R" dc lighting, and using the portable flashlights carried by station operators. A supply of flashlights, headlights, batteries and other lighting tools are routinely used by operators. The FLEX generators repower lighting panels at 2.5 hours from the initiation of the ELAP.

There are no emergency lighting fixtures in the outside areas of the protected area to provide necessary lighting in those areas where portable FLEX equipment is to be deployed. The F750 truck has hand-held flashlights that can be used while deploying FLEX equipment to further assist the staff responding to a BDBEE event during low light conditions. For additional flexibility, at each FLEX storage pad location, inside each sea van container, there are 5 LED stanchion lighting units and multiple lengths of extension cords that can be deployed to strategic locations if required. The LEDs can be powered by the Yanmar portable 5500W diesel generators stored in the sea van container.

For Phase 3, the NSRC is deploying portable lighting towers per the SAFER Response Plan. The deployment of three 30 ft. high, 440,000 lumens, and diesel-driven lighting towers from SAFER will support OCNGS exterior lighting for equipment staging areas and FLEX deployment locations.

3.7.5 Access to Protected and Vital Areas

In its FIP, the licensee stated that security doors and gates that rely on electric power to operate opening and/or locking mechanisms are barriers of concern. The security force will initiate an access contingency upon loss of power as part of the security plan. Access to the owner controlled area, site protected area, and areas within the plant structures will be controlled under this access contingency as implemented by security personnel.

During the audit process, the licensee provided information describing that access to protected areas will not be hindered. The licensee has contingencies in place to provide access to areas required for the ELAP response if the normal access control systems are without power.

3.7.6 Fueling of FLEX Equipment

In the FIP, Section 2.9.6 states that FO for the FLEX equipment will be obtained from the existing EDG FO storage tank. The tank is an above grade storage tank located inside the EDG building and the FO is accessible through two grade level valves. The staff noted that UFSAR Section 3.5.2 indicates that the Fuel Oil Tank Vault is a Seismic Class I structure that is protected from externally generated missiles. Based on the design of the EDG tank and its location within the EDG building, the staff finds the FO tank is robust and the FO contents should be available to support the licensee's FLEX strategies during an ELAP event.

The licensee explained that the TS required minimum quantity of FO in the EDG FO tank is 14,500 gallons. The licensee explained that the fuel consumption for FLEX portable equipment to support FLEX (one diesel driven portable pump, one diesel driven portable generator, and six small portable diesel driven generators) is approximately 13 gph, 34.4 gph, and 3.6 gph, respectively for a total of 51 gph for all FLEX equipment operating at full load. Based on this total fuel consumption rate for the FLEX equipment of 51 gph there is sufficient FO in the tank for 284 hours of operation (~11.8 days) without replenishment from outside sources. The staff finds it is a conservative assumption that the diesel-driven FLEX equipment will be operated continuously at full load because it would be expected that, as the reactor is cooled down, the demand on the equipment would decrease. Based on the conservative FO consumption rates and the protected FO volume on site, the staff finds the FO contents should be available to support the licensee's FLEX strategies during an ELAP event and that the quantity available is sufficient to support FLEX until off-site resources can provide FO replenishment to the site.

The licensee explained that its refueling FLEX equipment consists of two different methods for delivering FO to FLEX equipment. The first method consists of two 118-gallon fuel tanks mounted on the F750. The second method consists of a 100-gallon poly tank mounted on a pallet that is transported by the Kubota tractor to the EDG building. For either of these methods, a FO transfer rig that includes a portable 5500W diesel generator specific for the FO transfer rig, and two portable FO transfer hose reels (100' each). Once the FO transfer rig is staged, FO can be extracted from EDG FO tank to fill either the tanks mounted on the F-750 or the 100-gallon poly tank. For the first method, the F-750 transports the FO to its fill destination and has two 18 gpm electric fuel pumps, which are powered by the truck, (one for each FO tank) to transfer FO from the F-750 to the FLEX equipment (i.e. the FLEX pump, FLEX generator, and other miscellaneous users). For the second method, once this poly tank is filled, the tank is transported to the fill destination via the Kubota tractor. Then the FO is transferred out of the poly tank via an installed 12 Vdc pump (integral to the 100 gallon poly tank) powered by jumper cable clamps to the battery of the Kubota tractor. Based on the licensee's sequence of events, operators will begin operations to refuel portable FLEX equipment 14 hours after the initiating event. The FLEX generator is a trailer-mounted Cummins powered commercial generator with an integral 500-gallon fuel tank capable of supporting 14.5 hours of operation at full load and is expected to be in operation ~2.5 hours after the initiating event. The FLEX pump is a trailer-mounted Godwin DriPrime HL130M centrifugal pump with an integral 250 gallon fuel tank capable of supporting approximately 19 hours of operation at full load and is expected to be in operation ~100 minutes after the initiating event. Based on the available protected equipment to support refueling operation, the available run-time and FO consumption rate for each piece of FLEX equipment, the staff finds that diesel-powered FLEX equipment can be adequately refueled to ensure uninterrupted operation to support the licensee's FLEX strategies.

In the OCNGS UFSAR, Section A.1.22 indicates that an existing FO chemistry program monitors and controls FO contaminants in accordance with the guidelines of the American Society for Testing and Materials (ASTM) for the Emergency Diesel Generator Fuel Storage Tank. In addition, FIP Section 2.18.6 indicates that periodic testing and preventative maintenance of the FLEX equipment consistent with guidance from EPRI templates or manufacturer provided information/recommendations. The licensee explained that activities included periodic static inspections, fluid analysis, periodic operational verifications and periodic functional verifications. Based on the controls established in the TSs and preventative maintenance guidance from EPRI templates, the staff finds that the licensee has addressed management of FO quality in the EDG FO storage tank and portable FLEX equipment to ensure FLEX equipment will be supplied with quality FO during an ELAP event.

3.7.7 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should allow deploying the FLEX equipment following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01 and should adequately address the requirements of the order.

3.8 Considerations in Using Offsite Resources

3.8.1 Oyster Creek Generating Station SAFER Plan

The industry has collectively established the needed off-site capabilities to support FLEX Phase 3 equipment needs via the SAFER team. The SAFER team consists of the Pooled Equipment Inventory Company (PEICo) and AREVA Inc. and provides FLEX Phase 3 management and deployment plans through contractual agreements with every commercial nuclear operating company in the United States.

There are two NSRCs, located near Memphis, Tennessee and Phoenix, Arizona, established to support nuclear power plants in the event of a BDBEE. Each NSRC holds five sets of equipment, four of which will be able to be fully deployed to the plant when requested. The fifth set allows removal of equipment from availability to conduct maintenance cycles. In addition, the plant's FLEX equipment hose and cable end fittings are standardized with the equipment supplied from the NSRC.

By letter dated September 26, 2014 [Reference 21], the NRC staff issued its assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the staff concluded that SAFER has procured equipment, implemented appropriate processes to maintain the equipment, and developed plans to deliver the equipment needed to support site responses to BDBEEs, consistent with NEI 12-06 guidance; therefore, the staff concluded in its assessment that licensees can reference the SAFER program and implement their SAFER response plans to meet the Phase 3 requirements of Order EA-12-049.

The NRC staff noted that the licensee's SAFER Response Plan contains (1) SAFER control center procedures, (2) NSRC procedures, (3) logistics and transportation procedures, (4) staging area procedures, which include travel routes between staging areas to the site, (5) guidance for site interface procedure development, and (6) a listing of site-specific equipment (generic and non-generic) to be deployed for FLEX Phase 3.

3.8.2 Staging Areas

In general, up to four staging areas for NSRC supplied Phase 3 equipment are identified in the SAFER Plans for each reactor site. These are a Primary (Area C) and an Alternate (Area D), if available, which are offsite areas (within about 25 miles of the plant) utilized for receipt of ground transported or airlifted equipment from the NSRCs. From Staging Areas C and/or D, the SAFER team will transport the Phase 3 equipment to the on-site Staging Area B for interim staging prior to it being transported to the final location in the plant (Staging Area A) for use in Phase 3. For Oyster Creek, Staging Area C is the Atlantic City International Airport and Alternate Staging Area D is Ocean County Airport. Staging Area B is east side of site just on the other side of US Highway, Route 9.

Use of helicopters to transport equipment from Staging Area C and D to Staging Area B is recognized as a potential need within the Oyster Creek SAFER Plan and this option is addressed.

3.8.3 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should allow utilization of offsite resources following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01 and should adequately address the requirements of the order.

3.9 Habitability and Operations

3.9.1 Equipment Operating Conditions

3.9.1.1 Loss of Ventilation and Cooling

Following a BDBEE and subsequent ELAP event at Oyster Creek, ventilation that provides cooling to occupied areas and areas containing required equipment will be lost. Per the guidance given in NEI 12-06, FLEX strategies must be capable of execution under the adverse conditions (unavailability of installed plant lighting, ventilation, etc.) expected following a BDBEE resulting in an ELAP.

The primary concern with regard to ventilation is the heat buildup which occurs with the loss of forced ventilation in areas that continue to have heat loads. The licensee performed a loss of ventilation analyses to quantify the maximum steady state temperatures expected in specific areas of the plant related to FLEX implementation to ensure the environmental conditions remain acceptable and within equipment qualification limits.

The key areas identified for all phases of execution of the FLEX strategy activities are the main control room (MCR), reactor building (A/B battery room), and turbine building (480 Vac switchgear room, 4160 Vac switchgear and C battery room), and containment. The licensee evaluated these areas to determine the temperature profiles following an ELAP/LUHS event. The results of the licensee's room heat-up evaluations have concluded that temperatures remain within acceptable limits based on conservative input heat load assumptions for all rooms/areas using passive and active means of portable ventilation.

Main Control Room

The licensee performed calculation EXOC049-CALC-001, "Control Room FLEX Heat Up," Revision 0, which modeled the temperature transient response for 72 hours in the MCR and determined the actions that are needed to maintain equipment operability of the MCR following an ELAP. The calculation used the GOTHIC (Generation of Thermal-Hydraulic Information for Containments) version 8.1 computer program. The licensee assumed a maximum outside temperature of 106°F. The licensee's calculation determined that the maximum temperature in the MCR during the analysis would be 109.7°F at 1.5 hours following the ELAP event, at which time a portion of the MCR heat load is shed, decreasing the temperature to 87°F. The licensee also determined opening doors and the use of one 20-in. portable fan providing a minimum flow rate of 6700 cubic ft. per minute (cfm) is sufficient to maintain the MCR temperature. At 8 hours, the MCR temperature increases due to the portable fan starting and drawing in relatively warmer outside air at 106°F. The MCR temperature is expected to fluctuate with the change in temperature once the portable fan is started; however, the temperature is not expected to exceed 110°F throughout the event. Procedure FSG-15, "Establishing MCR and Battery Room Ventilation," Revision 0, provides guidance to run the portable fan before temperature reaches 104°F to maintain MCR temperature below 110°F during the ELAP event.

Based on temperatures remaining below 120°F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, for electronic equipment to be able to survive indefinitely), the NRC staff finds that the electrical equipment in the MCR will not be adversely impacted by the loss of normal ventilation as a result of an ELAP event

A/B Battery Room

The licensee performed calculation EXOC049-CALC-002, "A/B Battery Room FLEX Heat Up," Revision 0, which modeled the temperature transient response for 72 hours in the A/B Battery Room and determined the actions that are needed to maintain equipment operability following an ELAP event. The calculation used the GOTHIC version 8.1 computer program. The licensee's calculation determined that the A/B battery room temperature is expected to remain below 120°F with the actions of blocking open doors and powering up a 9500 cfm portable fan at approximately 14.5 hours after initiation of an ELAP event. The A/B battery room temperature is expected to fluctuate with the change in temperature once the portable fan is started, however the temperature is not expected to exceed 120°F throughout the event. Procedure FSG-15, provides guidance to run the 9500 standard CFM portable fan before temperature reaches 114°F to maintain A/B battery room temperature below 120°F during the ELAP event.

Based on the above, the NRC staff finds that the licensee's ventilation strategy (opening doors and portable ventilation) should maintain the battery room temperature below the maximum temperature limit specified by the manufacturer (less than 122°F for short durations) of the B station batteries, therefore the batteries should not be adversely impacted by the loss of ventilation as a result of an ELAP event. Although the licensee plans to open doors and provide portable ventilation, periodic monitoring of electrolyte level may be necessary to protect the battery since the battery may gas more at higher temperatures.

480 Vac Switchgear Rooms

The licensee performed calculation EXOC049-CALC-003, "480V Switchgear Room FLEX Heat Up," Revision 0, which modeled the temperature transient response for 72 hours in the 480 Vac Switchgear A and B Rooms and determined the actions that are needed to maintain equipment operability following an ELAP event. The licensee's calculation used the GOTHIC version 8.1 computer program. Based on the 480 Vac Switchgear Room GOTHIC calculation, the licensee's strategy is to open doors at approximately 10 hours after the ELAP. The licensee's calculation determined that by opening doors, the temperatures in the 480 Vac Switchgear A and B Rooms would remain below the acceptance criteria of 120°F over the 72-hour transient time.

Based on temperatures remaining below 120°F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1), the NRC staff finds that the electrical equipment in the 480 Vac switchgear rooms will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

4160 Vac Switchgear Rooms and C Battery Room

The licensee performed calculation EXOC049-CALC-004, "4160 and Battery Room C Heat Up for Beyond Design Basis External Event with Extended Loss of AC Power," Revision 0, which modeled the temperature transient response for 72 hours in the 4160 Vac Switchgear A/B

Rooms and C Battery Room, and determined the actions that are needed to maintain equipment operability following an ELAP event. The licensee's calculation used the GOTHIC version 8.1 computer program. The licensee's calculation determined that without taking any operator actions, neither the 4160 Vac Switchgear A/B Room nor the C Battery Room exceed the acceptance criteria of 120°F for 72 hours. The diurnal outside temperature along with a low heat load in the 4160 Vac Switchgear A/B Room effectively mitigates the temperature rise in these rooms. The 4160 Vac Switchgear A/B Room and C Battery Room temperature is expected to reach a maximum of 115°F and 118°F at 72 hours, respectively. Although the licensee calculation determined that no mitigating actions are required to maintain temperature below 120°F, procedure FSG-15, provides guidance to restore C Battery Room ventilation within 30 hours to minimize the buildup of hydrogen which should also assist in maintaining temperature below 120°F.

Based on temperatures remaining below 120°F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1), the NRC staff finds that the electrical equipment in the 4160 Vac Switchgear A/B Room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Based on the above, the NRC staff finds that if the C battery room temperature remains below the maximum temperature limit (120°F) of the station batteries, as specified by the battery manufacturer (Exide Technologies), the loss of ventilation as a result of an ELAP event should not adversely impacted the station batteries. Although the licensee plans to restore battery room ventilation, periodic monitoring of electrolyte level may be necessary to protect the battery since the battery may gas more at higher temperatures.

FLEX Equipment Operated Outdoors

In the FIP, Section 2.6.5 stated that portable diesel equipment and portable static battery chargers will be operated in areas outdoors, where the temperature will effectively be ambient. The licensee confirmed that based on vendor information the equipment is expected to operate at the extreme high temperatures for the site. The FLEX pump and FLEX generator will be stored outdoors and the licensee confirmed that the equipment will be maintained at a temperature, which ensures functionality when called upon.

Containment

The licensee performed calculation OC-MISC-010, "MAAP Analysis to Support FLEX Initial Strategy," Revision 1, which modeled the transient temperature response in the containment following an ELAP event. The calculation analyzed the containment pressure and temperature response for 72 hours following an ELAP. The results of this analysis determined that by isolating containment and utilizing the IC, the containment design limits for pressure (35 psig) and temperature (281°F) are not challenged during this period. The ICs should automatically initiate at the beginning of an ELAP, and the operators' should be able to achieve a ≤10°F/hour cooldown rate. Once the potential ELAP condition has been declared, the RPV cooldown rate will increase to 50°F/hour. The maximum drywell pressure and temperature reached is 15.1 psig and 261°F during the 72-hour period. The maximum suppression pool pressure and temperature reached is 13.5 psig and 98°F during the 72-hour period. The containment pressure will be monitored and, if necessary, the containment will be vented using existing procedures and systems. Based on containment temperature and pressure remaining below their respective design limit and the ability to vent the containment, the NRC staff expects that the necessary equipment, including credited instruments, located inside containment should remain functional throughout an ELAP event.

Based on its review of the essential station equipment required to support the FLEX mitigation strategy, which are primarily located in the MCR, reactor building (A/B battery room), turbine building (480 Vac switchgear room, 4160 Vac switchgear and C battery room), and containment, the NRC staff finds that the equipment should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP/LUHS event.

3.9.1.2 Loss of Heating

In the FIP, Section 2.6.4 states that when FLEX equipment is stored outdoors on the FLEX Storage Pad, the licensee will provide power to supply direct heating (e.g., jacket water, battery, engine block heater, etc.) in addition to using a starting battery trickle charger to maintain the batteries at full charge for cold weather starts. The licensee also confirmed that the diesel fuel for FLEX equipment is treated with a fuel additive during cold weather conditions to prevent gelling; thus, ensuring the FLEX equipment will operate as expected.

In addition, during FLEX pump operation the licensee explained in FIP Section 2.3.9 that the constant flow line on the FLEX manifold must be aligned maintained with flow to prevent freezing during extreme cold temperatures. Furthermore, during its audit, the staff reviewed the procedural guidance for the FLEX pump operation and noted that operators are providing warnings to establish recirculation flow external to the pump or disconnecting/draining hoses when the pump is not in use. Based on the procedural guidance to drain hoses when not in use during extreme cold temperatures or establish recirculation flow external to the pump, the staff finds it reasonable that extreme low temperatures will not impact the FLEX pump operation and the licensee's FLEX strategies.

The Oyster Creek Class 1E station battery rooms are located inside the reactor building and turbine building, and would not be exposed to extreme low temperatures. At the onset of the event, the battery rooms would be at their normal operating temperature and the temperature of the electrolyte in the cells would build up due to the heat generated by the batteries discharging and during re-charging. In calculation EXOC049-CALC-002, the licensee determined that the A/B Battery Room temperature does not drop below the acceptance criteria of 60°F for the extreme cold case as long as doors are not opened during the 72-hour transient. In calculation EXOC049-CALC-004, the licensee determined that the C Battery Room temperature does not drop below the acceptance criteria of 50°F over the 72-hour transient in the extreme cold case. The temperature is expected to reach approximately 51°F at 72 hours. The resultant temperature of 51°F is the ambient air in the room; however the TSs require the licensee to monitor the temperature of the electrolyte, which would be at a higher temperature than the room itself due to the internal generation of heat from the batteries' operation. Therefore, the NRC staff finds that Oyster Creek station batteries should perform their required functions as a result of loss of normal heating during an ELAP event.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3, is the potential buildup of hydrogen in the battery rooms as a result of loss of ventilation during an ELAP event. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging.

The NRC staff reviewed licensee calculation C-1302-735-E320-050, "Oyster Creek Batteries A, B, C Hydrogen Calculation," Revision 0. The licensee's calculation showed that 1 percent hydrogen concentration would be reached in the A/B and C battery room within 87.4 and 31.2 hours, respectively without any ventilation. For the A/B battery room, procedure FSG-15 provides guidance to block open doors and restore battery room ventilation using a portable fan within 86 hours. For the C battery room, procedure FSG-15 provides guidance to restore normal ventilation to the 4160 V room within 30 hours.

Based on its review of the licensee's battery room hydrogen calculation and ventilation strategy, the NRC staff finds that hydrogen accumulation in the Oyster Creek vital battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP as a result of a BDBEE.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

As described above in Section 3.9.1.1, the MCR temperature profiles were determined in a calculation that used the GOTHIC code. The licensee explained that for the two cases investigated in its calculation, a maximum outside ambient temperature of 106 °F was used, which is the maximum-recorded temperature at the site. For Case 1, if no mitigating actions are taken following an ELAP event, the MCR temperature exceeds 110 °F at approximately 41 hours. For Case 2, the use of one portable fan 8 hours after the initiating event will result in the MCR temperature reaching 97 °F approximately 72 hours after the initiating event. During its audit, the staff noted that procedural guidance is provided to operators in procedure 331.1 "Control Room and Old Cable Spreading Room Heating, Ventilation and Air Conditioning System," Revision 40, to establish temporary ventilation for the MCR prior to exceeding 104 °F.

Based on the expected temperature response in the MCR and the procedural guidance to establish temporary ventilation to ensure temperatures remain below the limit of 110 °F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, for control room habitability), the staff finds it reasonable that operators can safely enter and occupy the MCR during an ELAP event. The staff finds the above strategies are consistent with NEI 12-06, Section 3.2.2 such that station personnel can safely enter and perform the necessary actions to support the FLEX mitigation strategy, during an ELAP event.

3.9.2.2 Spent Fuel Pool Area and Reactor Building

In the FIP, Section 2.5.4.3 indicates that during an ELAP/LUHS event the normal and emergency reactor building ventilation and normal cooling to the spent fuel pool will not be functional. The gradual heat up of the spent fuel pool and the two ICs when placed in service will generate heat in the reactor building.

The licensee performed an evaluation to identify any temperature limitations for accessibility on the SFP refueling deck during an ELAP event. The evaluation determined that the spent fuel pool refueling deck will reach a peak temperature of 109 °F and limit operator stay-time access to 55 minutes for moderately heavy work for up to 8 hours into the ELAP event.

The SFP cooling FLEX strategies require operator access to the reactor building 75-ft. elevation to connect FLEX hose to the SFP diffuser piping and to the reactor building 119-ft. elevation to deploy FLEX hoses/spray nozzles to the edge of the SFP. The licensee explained that a natural ventilation draft is created since the reactor building elevations (95-ft, 75-ft, 51-ft, and 23-ft) open to the reactor building 119-ft. elevation through the northwest stairway and the reactor building equipment hatch and eventually to the atmosphere via the reactor building 119-ft. elevation airlock manway. Thus, the staff finds it reasonable that the maximum temperature and stay-time determined by the licensee's evaluation for the SFP refueling deck bound the maximum temperatures and stay times for the lower reactor building elevations where other FLEX activities occur for deploying hoses and cables.

During its audit, the staff noted that procedural guidance is provided to operators to deploy hoses to reactor building elevations 119-ft. and 75-ft. within 6 hours of the initiating event prior to SFP boiling. In addition, the staff noted that procedural guidance directs operators to open specified doors/hatches in the reactor building to establish passive ventilation and that cautions are provided to establish ventilation prior to habitability concerns. Furthermore, hose deployment for IC makeup and RPV makeup occur in advance of SFP boiling in order to the support the mission times of 100 minutes and 3.3 hours, respectively.

Based on the procedural guidance to deploy hoses and open the necessary doors/hatches in the reactor building prior to the SFP becoming uninhabitable due to bulk boiling, the staff finds the above strategies are consistent with NEI 12-06, Section 3.2.2, such that station personnel can safely enter the reactor building (including the SFP refuel floor) and perform the necessary actions to support the FLEX mitigation strategy, during an ELAP event.

3.9.2.3 Other Plant Areas

In its FIP, the licensee stated that Oyster Creek onsite operation's personnel have on-hand cold weather garments and rain gear, in various sizes, for responders to wear during foul weather conditions. This same foul weather gear will be used to support outside FLEX deployment actions. All foul weather gear for the operators is stored in a locker room area adjacent to the MCR.

3.9.3 Conclusions

The NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain or restore equipment and personnel habitability conditions following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.10 Water Sources

Condition 3 of NEI 12-06, Section 3.2.1.3, states that cooling and makeup water inventories contained in systems or structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles are available. OCNGS has chosen to use intake/discharge canal fed by waters from Barnegat Bay as the water source throughout the ELAP/LUHS event. Barnegat Bay is a large natural body of water connected to the Atlantic Ocean and is the UHS for the plant. As such, an unlimited amount of water is available to cope with the ELAP event indefinitely. This allows the FLEX strategy to position the FLEX pumps at a source that can provide make-up water to the reactor pressure vessel, ICs, and spent fuel pool that is unlimited.

3.10.1 RPV Make-Up, IC Make-Up and Spent Fuel Pool Make-Up

In the FIP, Section 2.15 states that the intake/discharge canal water from Barnegat Bay is the primary water source throughout the ELAP event since the FLEX Pump can be staged at this source and provide an unlimited amount of makeup water to the RPV, ICs, and the SFP. The licensee explained that an evaluation was completed that verified that the necessary flowrates to support the FLEX strategies can be obtained by taking suction from the intake/discharge canal water from Barnegat Bay with a suction strainer at the end of the suction hose.

The primary suction point is downstream of the Intake Grates, Traveling Screens, Circulating Water Pumps, and the Main Condensers at the Circulating Water Pump discharge tunnel chemistry sample point, which will filter the water from the normal flow process stream. The licensee explained that with a loss of power there will not be any flow from the normal pumps and the only flow will be from the FLEX Pump suction. The maximum FLEX Pump suction flow rate will be less than 500 gpm compared to the normal flow rate of approximately 500,000 gpm from the Circulating Water Pumps, Service Water Pumps, and other smaller site support intake pumps. The staff noted that this comparatively lower FLEX flowrate through the large intake flow area results in the intake water flowing at a very low velocities, which will assist in reducing debris drawn into the suction strainer attached to the FLEX Pump suction hose. The alternate suction point for the FLEX Pump is available from the west side of the Circulating Water Pumps. Access to this suction is from the circulating water system intake tunnel and requires repositioning deck plate access openings in between the Intake Traveling Screens and the Circulating Water Pumps. The 6" suction hose can be routed from the FLEX Pump suction through the opening in the deck plate to the water source where water will be drawn through a suction strainer to further limit solid debris to prevent potential damage to the FLEX Pump.

The FLEX Pump configuration when using the primary or alternate suction point includes the use of 6-in. suction hose for the FLEX Pump attached to a barrel-shaped strainer, which consists of 1-in. holes to prevent larger debris from becoming entrained in the pump suction line that is submerged below the water surface. There are up to six sections of 10-ft. suction hose available and procedural guidance provides the flexibility to install the necessary number of suction hose such that the barrel strainer will be submerged below the water's surface and above the bottom of the Circulating Water Pump tunnel. The staff noted that the FLEX Pump will not begin operating until approximately 100 minutes after the initiation event, at which time it is expected that any heavier debris in the tunnel would settle toward the bottom and lighter debris would float toward the surface.

Furthermore, during its audit, the staff reviewed the licensee's available procedural guidance and noted that the control room is directed to establish a clean water source of makeup to the RPV and ICs. The available options are to determine the condition of the CST and start a CRD pump, contact the Emergency Response Organization (ERO) to procure a Salt Water Filtration system to be connected to the FLEX Pump or have the ERO evaluate connecting the FLEX Pump to the CST, Demineralized Water Tank or Redundant Fire Pump. Consistent with NEI 12-06, Section 3.2.2, the licensee's procedures/guidance include actions to be taken to transition to a more preferable water source as soon as is practical; thus, a heat transfer analysis is not necessary for using raw water in its FLEX strategies.

The staff finds it reasonable that the water quality from the intake/discharge canal water will not impact the implementation of the licensee's FLEX strategy because of the FLEX Pump suction point locations, the use of a suction strainer, and the ability to draw makeup water from a depth where debris is limited. The licensee determined that debris would not block flow through the core spray system. Further, because the licensee is using the core spray system for RV injection, it reduces concerns that the use of raw water will result in debris-blockage of the core inlet that could impede core cooling. In addition, the staff finds that the licensee has sufficient time to assess potential clean water sources for injection to the RPV, ICs and SFP, and establish the ability to filter the salt water from the intake/discharge canal water. The staff noted that due to the lack of fully robust water tanks located onsite and the time constraint to provide makeup to the ICs, the licensee's FLEX strategy relies on the intake/discharge canal water even though the water quality is not very high, since its availability is assured due to its robust design with respect to the external hazards.

3.10.2 Containment Cooling

The licensee stated that analysis shows that the containment does not exceed its design pressure or temperatures limits and as such, containment cooling water is not needed to cope with the ELAP event.

3.10.3 Conclusions

Based on the evaluation above, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain satisfactory water sources following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01 and should adequately address the requirements of the order.

3.11 Shutdown and Refueling Analyses

Order EA-12-049 requires that licensees must be capable of implementing the mitigation strategies in all modes. In general, the discussion above focuses on an ELAP occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. When the ELAP occurs with OCNGS at power, the mitigation strategy initially focuses on the use of the ICs to provide decay heat removal. Within 100 minutes of the ELAP, the licensee expects to establish makeup to the shell side of the ICs using a diesel driven FLEX pump. This time frame will ensure that the condenser tubes remain covered to maintain decay heat removal. Section 2.16 of the OCNGS FIP states that the FLEX strategies for Cold Shutdown are the same as those for Power Operation, Startup, and Hot Shutdown. Per Section 2.5.2 of the licensee's FIP, the OCNGS analysis shows that under the maximum SFP design-basis heat load, about 35 hours are available to implement makeup before the SFP level reaches 10 ft above the spent fuel (the level below which is assumed to prohibit access to the refuel floor from a radiological perspective). The licensee has stated that they have the ability to implement makeup to the SFP within that time.

If the plant is in Refueling mode with the RPV head removed and all or most of the fuel has been removed from the RPV and placed in the SFP, there is a shorter timeline to implement the makeup of water to the SFP. However, this is balanced by the fact that if immediate cooling is not required for the fuel in the reactor vessel, the operators can concentrate on providing makeup to the SFP. In this circumstance, the ICs are not available and should an ELAP occur, transition directly to FLEX Phase 2 must be initiated within 10 minutes as directed in Section 4.1(3) of procedure ABN-36, "Loss of Offsite Power & Station Blackout."

The NRC-endorsed strategy is described in NEI 12-06. Section 3.2.3 provides guidance to licensees for reducing shutdown risk by incorporating FLEX equipment in the shutdown risk process and procedures. Considerations in the shutdown risk assessment process include maintaining necessary FLEX equipment readily available and potentially pre-deploying or pre-staging equipment to support maintaining or restoring key safety functions in the event of a loss of shutdown cooling. In its FIP, the licensee stated that it would follow this guidance. During the audit process, the NRC staff observed that the licensee had made progress in implementing this guidance.

Based on the licensee's incorporation of the use of FLEX equipment in the shutdown risk process and procedures, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately should maintain or restore core cooling, SFP cooling, and containment following a BDBEE in shutdown and refueling modes consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.12 Procedures and Training

3.12.1 Procedures

In its FIP, the licensee described that the inability to predict actual plant conditions that require the use of BDB equipment makes it impossible to provide specific procedural guidance. As such, the FSGs provide guidance that can be employed for a variety of conditions. Clear criteria for entry into FSGs will ensure that FLEX strategies are used only as directed for BDBEE conditions, and are not used inappropriately in lieu of existing procedures. When FLEX equipment is needed to supplement EOPs or severe accident mitigation guidelines (SAMGs), those procedures will direct the entry into and exit from the appropriate FSG procedure.

In its FIP, the licensee stated that FLEX strategy support guidelines have been developed in accordance with BWR owners group (BWROG) guidelines. The FSGs will provide available, preplanned FLEX strategies for accomplishing specific tasks in the EOPs or SAMGs. The FSGs will be used to supplement (not replace) the existing procedure structure that establishes command and control for the event.

3.12.2 Training

In its FIP, the licensee stated that the Nuclear Training Program has been revised to assure personnel proficiency in the mitigation of BDBEEs is adequate and maintained. These programs and controls were developed and have been implemented in accordance with the Systematic Approach to Training (SAT) process.

The licensee stated that initial training has been provided and periodic training will be provided to site emergency response leaders on BDBEE emergency response strategies and implementing guidelines. Personnel assigned to direct the execution of mitigation strategies for BDBEEs have received the necessary training to ensure familiarity with the associated tasks, considering available job aids, instructions, and mitigating strategy time constraints. Where appropriate, integrated FLEX drills will be organized on a team or crew basis and conducted periodically with all time-sensitive actions to be evaluated over a period of not more than eight years. It is not required to connect/operate temporary/permanently installed equipment during these drills.

3.12.3 Conclusions

Based on the description above, the NRC staff finds that the licensee has adequately addressed the procedures and training associated with FLEX. The procedures have been issued in accordance with NEI 12-06, Section 11.4, and a training program has been established and will be maintained in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 [Reference 40], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 41], the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs. In its FIP, the licensee stated that they would conduct maintenance and testing of the FLEX equipment in accordance with the industry letter.

In its FIP, the licensee stated that EPRI templates are used for FLEX equipment where applicable. However, in those cases where EPRI templates were not available, preventative maintenance actions were developed based on manufacturer provided information and recommendations and Exelon fleet procedure ER-AA-200, "Preventive Maintenance Program".

The NRC staff finds that the licensee has adequately addressed equipment maintenance and testing activities associated with FLEX equipment because a maintenance and testing program has been established in accordance with NEI 12-06, Section 11.5.

3.14 Alternatives to NEI 12-06, Revision 2

The NRC staff identified the licensee's use of the FLEX manifolds as an alternative from the guidance in NEI 12-06, Revision 2, Section 3.2.2, to have primary and alternate connections for the FLEX strategies; specifically, the licensee's ability to supply water to the RPV, ICs, and SFP is entirely dependent on the FLEX manifolds. Based on the robust design/installation and location of the FLEX manifolds, and the licensee's evaluation on potential seismic interaction between the manifolds and adjacent components, the NRC staff determined that the FLEX manifolds, including the riser piping, are reasonably protected from all applicable external hazards. (See Section 3.7.3.1 of this SE.) Therefore, the NRC staff approves this alternative as being an acceptable method of compliance with the order.

In conclusion, the NRC staff finds that although the guidance of NEI 12-06 has not been met, if the above alternative is implemented as described by the licensee, the requirements of the order will be met.

3.15 Conclusions for Order EA-12-049

Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance to maintain or restore core cooling, SFP cooling, and containment following a BDBEE which, if implemented appropriately, should adequately address the requirements of Order EA-12-049.

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 22], the licensee submitted its OIP for OCNGS in response to Order EA-12-051. By letter dated August 28, 2013 [Reference 23], the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response by letter dated September 18, 2013 [Reference 24]. By letter dated November 8, 2013 [Reference 25], the NRC staff issued an Interim Staff Evaluation (ISE) and RAI to the licensee.

By letters dated October 25, 2012 [Reference 26], August 28, 2013 [Reference 27], February 28, 2014 [Reference 28], August 28, 2014 [Reference 29], February 27, 2015 [Reference 30], August 28, 2015 [Reference 31], February 26, 2016 [Reference 32] and August 26, 2016 [Reference 33], the licensee submitted responses for the ISE, RAI and status reports for the Integrated Plan. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable SFP level instrumentation, which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated December 1, 2016 [Reference 35 (Non-Public)], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed an SFP level instrumentation system designed by Westinghouse. The NRC staff reviewed the vendor's SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report on August 18, 2014 [Reference 34].

The staff performed an onsite audit to review the implementation of SFP level instrumentation related to Order EA-12-051. The scope of the audit included verification of (a) site's seismic and environmental conditions enveloped by the equipment qualifications, (b) equipment installation met the requirements and vendor's recommendations, and (c) program features met the requirements. By letter dated August 29, 2016 [Reference 17], the NRC issued an audit report on the licensee's progress. Refer to Section 2.2 above for the regulatory background for this section.

4.1 Levels of Required Monitoring

In a letter dated August 26, 2016 [Reference 33], the license stated that:

> Reference Attachment 1 for a sketch depicts the elevation view of the proposed mounting arrangement of the mounting brackets and level sensors. Datum points Level 1, Level 2, Level 3, top of fuel racks and level sensor measureable range are all depicted in this sketch.
>
> LEVEL 1 - This is the level at which point the reliable suction loss occurs due to uncovering of the weirs. Level 1 corresponds to elevation 117 ft. 10 in.
>
> LEVEL 2 - This is the level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. Level 2 corresponds to elevation 106 ft. 1 1/8 in. water level 10 ft. above the highest point of any fuel rack seated in the spent fuel pool.
>
> LEVEL 3 - This is the level where fuel remains covered and actions to implement make-up water addition may no longer be deferred. This level corresponds to a water level 7 7/8 in. above the top of the fuel assembly. Level 3 shall be designated at elevation 96 ft. 9 in.

The NRC staff found the licensee selection of the SFP measurement level adequate based on the following:

- Level 1 is adequate for normal SFP cooling system operation, and it is also adequate to ensure the required fuel pool cooling pump net positive suction head as the skimmer surge tanks supply the SFP cooling pumps. This level represents the higher of the two points described in NEI 12-02 for Level 1.

- Level 2 meets the first option described in NEI 12-02, which is 10 ft. (+/- 1 ft.) above the highest point of any fuel rack seated in the SFP. The designed Level 2 represents the range of water level where any necessary operations in the vicinity of the SFP can be completed without significant dose consequences from direct gamma radiation from the SFP consistent with NEI 12-02.

- Level 3 is above the highest point of any fuel storage rack seated in the SFP. The level allows the licensee to initiate water make-up with no delay meeting the NEI 12-02 specifications of the highest point of the fuel racks seated in the SFP. Meeting the NEI 12-02 specifications of the highest point of the fuel racks conservatively meets the Order EA-12-051 requirement of a level where the fuel remains covered.

Based on the evaluation above, the NRC staff finds that the licensee's proposed Levels 1, 2 and 3 appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2 Evaluation of Design Features

Order EA-12-051 required that the SFP level instrumentation shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Refer to Section 2.2 above for the requirements of the order in regards to the design features. Below is the staff's assessment of the design features of the SFP level instrumentation.

4.2.1 Design Features: Instruments

By letter dated August 26, 2016 [Reference 33], the licensee stated that the two channels of the proposed level measurement system will be installed such that the level probes will be mounted on the south side of the SFP and will be separated by a distant greater than the span of the shortest side of the pool. The level sensors' measurement range is between elevation 96 ft. 9 in. and 118 ft. 6 7/8 in.

The NRC staff notes that the range specified for the licensee's instrumentation will cover Levels 1, 2, and 3, as described in Section 4.1 above.

The NRC staff finds that the licensee's design, with respect to the number of channels and measurement range for its SFP level instrumentation, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.2 Design Features: Arrangement

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

> The Reactor Building [RB], Control Room [CR], Cable Tray Bridges [CTB], Lower Cable Spreading Room [LCSR] and Upper Cable Spreading Room [UCSR] (in which the equipment and cabling are located) are seismic I structures and will remain operable during and after a beyond design basis event. All equipment and cabling are installed to achieve maximum practical separation in accordance with NEI 12-02 Revision 1 requirements while meeting 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. In the RB refuel floor area, the two sensors will be mounted in different locations of the SFP and separately by a distance comparable to the shortest side of the pool. The coaxial cables are protected in conduit and are in the refuel floor for a short distance. The coaxial cables that extend from the two sensors toward the location of the transmitters (sensors electronics) have a minimum separation of 8 ft., exceeding the 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. The primary coaxial cable will be installed in conduit and the secondary coaxial cable will be installed in combination of conduit and cable tray. The primary transmitter is located in the south CTB and the secondary transmitter is located 25 ft. away in the north CTB. The electronic enclosures (displays) located in the UCSR are mounted 5 ft. apart. The 4-20 mA cables that extend from the transmitters to the displays have a minimum separation of 1 in., maintaining the 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. The primary 4-20 mA cables will be installed in combination of

> conduit and cable tray. Distribution panels PDP-733-57 & PDP-733-58 are located in the LCSR. The 120 VAC cables from the distribution panels terminate to panel 17R in the CR and to the displays in the UCSR. The 120 VAC power cables are 1 in. apart maintaining the 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. All the equipment and supports will be seismically mounted.

The NRC staff noted that there is sufficient channel separation between the primary and back-up level instruments, sensor electronics, and routing cables to provide reasonable protection against loss of indication of SFP level due to missiles that may result from damage to the structure over the SFP.

Based on the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee's proposed arrangement for the SFP level instrumentation appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.3 <u>Design Features: Mounting</u>

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

> a) All SFPIS [Spent Fuel Pool Instrumentation System] equipment is designed in accordance with the Oyster Creek Safe Shutdown Earthquake (SSE) design requirements.
>
> The vendor, Westinghouse, has evaluated the structural integrity of the mounting brackets in calculations CN-PEUS-15-08. The GTSTRUDL model, used by Westinghouse to calculate the stresses in the bracket assembly, considered load combinations for the dead load, live load and seismic load on the bracket. The reactionary forces calculated from these loads become the design inputs to design the mounting bracket anchorage to the refuel floor to withstand a Safe Shutdown Earthquake (SSE).
>
> Seismic
>
> The seismic loads are obtained from the response spectra curves of Oyster Creek. The following methodology was used in determining the stresses on the bracket assembly:
>
> - Frequency analysis, taking into account the dead weight and the hydrodynamic mass of the structure, is performed to obtain the natural frequencies of the structure in all three directions.
> - SSE (Safe Shutdown Earthquake) response spectra analysis is performed to obtain member stresses and support reactions.
> - Modal responses are combined using the Ten Percent Method per U.S. NRC Regulatory Guide 1.92, Revision 1, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".
> - The seismic loads for each of the three directions are combined by the Square Root of the Sum of Squares (SRSS) Method.

- Sloshing analysis is performed to obtain liquid pressure and its impact on bracket design.
- The seismic results are combined with the dead load results and the hydrodynamic pressure results in absolute sum. These combined results are compared with the allowable stress values.

Sloshing

Sloshing forces were obtained by analysis. The TID-7024, Nuclear Reactors and Earthquakes, 1963, by the US Atomic Energy Commission, approach has been used to estimate the wave height and natural frequency. Horizontal and vertical impact force on the bracket components was calculated using the wave height and natural frequency obtained using TID-7024 approach. Using this methodology, sloshing forces have been calculated and added to the total reactionary forces that would be applicable for bracket anchorage design. The analysis also determined that the level probe can withstand a credible design basis seismic event. During the design basis event, the SFP water level is expected to rise and parts of the level sensor probe are assumed to become submerged in water. The load impact due to the rising water and submergence of the bracket components has also been considered for the overall sloshing impact. Reliable operation of the level measurement sensor with a submerged interconnecting cable has been demonstrated by analysis of previous Westinghouse testing of the cable, and the vendor's cable qualification.

The following Westinghouse documents provide information with respect to the design criteria used, and a description of the methodology used to estimate the total loading on the device.

1) Calculation CN-PEUS-15-08, "Seismic Analysis of the SFP Primary and Backup Mounting Brackets at Oyster Creek Nuclear Generating Station"
2) EQ-QR-269, WNA-TR-03149-GEN Seismic Qualification of other components of SFPIS.

Oyster Creek Station specific evaluations were performed in Engineered Change Request (ECR) OC 14-00389 to address the seismic qualification and structural requirements of other SFPIS equipment. The design criteria in the ECR meet the requirements to withstand a SSE. The methods used in the ECR follow SQUG-GIP, Revision 2A, IEEE Standard 344-2004 and IEEE Standard 323-2003 for seismic qualification of the instrument.

(b) The level sensor, which is one long probe, will be suspended from the launch plate via a coupler/connector assembly. The launch plate is a subcomponent of the bracket assembly, which will be mounted on the refueling floor (Reactor Building EL. 119 ft.-3 in.) via concrete anchors. Drawing 10067E58, Sheet 1 prepared by Westinghouse shows the connection/mounting details.

(c) The bracket assembly that supports the sensor probe and launch plate will be mechanically fastened to the concrete slab of the refueling floor. Each mechanical connection consists of four concrete expansion anchors that will bolt

> the bracket assembly to the SFP structure via a base plate. The concrete expansion anchors will be designed to withstand SSE and will meet the Oyster Creek safety related installation requirements. The analysis and details of the Pool-side brackets are provided in Calculation CN-PEUS-15-08 and Drawing 10067E58, Sheet 1. The design and details of the anchorage to the floor are provided in ECR OC 14-00389.

During the onsite audit, the NRC staff reviewed the mounting specifications and seismic analyses for the SFP level instrumentation (SFPLI), including the methodology and design criteria used to estimate the total loading on the mounting devices. The staff also reviewed the design inputs and the methodology used to qualify the structural integrity of the affected structures for each of the SFPLI mounting attachments. Based on the review, the staff found the criteria established by the licensee adequately account for the appropriate structural loading conditions, including seismic and hydrodynamic loads.

Based on the evaluation above, the NRC staff finds that the licensee's proposed mounting design appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

Appendix A-1 of the guidance in NEI 12 02 describes a quality assurance process for non-safety systems and equipment that are not already covered by existing quality assurance requirements. In JLD-ISG-2012-03, the NRC staff found the use of this quality assurance process to be an acceptable means of meeting the augmented quality requirements of Order EA-12-051.

In its OIP, the licensee stated that instrument channel reliability shall be established by use of an augmented quality assurance process similar to that applied to the site fire protection program.

The NRC staff finds that, if implemented appropriately, this approach appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4.2 Instrument Channel Reliability

Section 3.4 of NEI 12-02 states, in part:

> The instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and/or testing of channel components for the following sets of parameters, as described in the paragraphs below:
>
> - conditions in the area of instrument channel component use for all instrument components,

- effects of shock and vibration on instrument channel components used during any applicable event for only installed components, and
- seismic effects on instrument channel components used during and following a potential seismic event for only installed components.

Equipment reliability performance testing was performed to (1) demonstrate that the SFP instrumentation will not experience failures during BDB conditions of temperature, humidity, emissions, surge, and radiation, and (2) to verify those tests envelope the plant-specific requirements.

The NRC staff reviewed the Westinghouse SFPLI's qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic [Reference 34]. The staff further reviewed the anticipated OCNGS environmental condition during the onsite audit [Reference 17]. Below is the staff's assessment of the equipment reliability of Oyster Creek SFP level instrumentation.

4.2.4.2.1 Radiation, Temperature, and Humidity

In a letter dated August 26, 2016 [Reference 33], the licensee stated that:

> Environmental Conditions for SFPIS components installed in the Spent Fuel Pool area at Oyster Creek are bounded by Westinghouse test conditions. Westinghouse qualified SFPLI to operate in BDB conditions for radiation TID [total integrated dose] 1.E07 Rads γ. During the BDB event with water at Level 3 (12" above top of fuel rack), at Oyster Creek [Oyster Creek] station radiation TID at coupler connection will be 8E06 Rads γ (per Calculation BYR13-051 - NEI 12-02 Spent Fuel Pool Doses). The radiation value of 8E06 R γ is lower than 1.E07 R γ to which Westinghouse qualified the coupler connection and as a result it is bounded. Below the coupler connection, the probe is made of stainless steel. Stainless steel is resistant to radiation, therefore, there is no concern with the SFPI operation (per Westinghouse Letter LTR-SFPIS-13-35). The equipment is qualified by Westinghouse to function during BDB events at 212 deg F and 100% humidity conditions to meet NEI 12-02 Revision 1 requirements (EQ-QR-269). As a result, SFPI is qualified to function under BDB environmental conditions of heat, humidity and radiation at Oyster Creek in the SFP area.
>
> Mild Environment – Westinghouse qualified the system components (display panel, transmitter) that reside in the mild environment conditions to determine that the components can satisfactorily perform to those conditions. Westinghouse has determined that aging does not have a significant effect on the ability of the equipment to perform following a plant design basis earthquake. Exelon has reviewed the documents and found them acceptable. Reference Westinghouse documents EQ-QR-269, WNA-TR-03149-GEN for description of specific qualification methods.

Below are specific details for OYS equipment qualifications:

Transmitter

Radiation: Oyster Creek Environmental Qualification report ES-027 Revision 4 states radiation limit "Not Available" where the transmitter is installed. The area is a non-harsh environment. Per the radiological survey map for this location during normal operation, the radiation in this area (Cable Tunnel above Office Room and Turbine Building) will be <1 mR/hr, and as a result is not expected to exceed 1.E03 Rads.

Temperature: Per SDD-OC-811-A DIV1, Rev 1, Fire Protection New Cable Spreading Room & Cable Bridge Tunnel document, the cable bridge tunnel area, where the transmitter will be installed, can experience temperature between 0 - 130 deg F. For BDB conditions, transmitters are qualified up to 140 deg F. As a result, there is no concern with operating transmitters in cable bridge tunnel area during BDB event conditions. During normal operating conditions, transmitters are qualified by Westinghouse to operate between 50 deg F - 120 deg F per EQ-QR-269. However, upon further investigation, Westinghouse found transmitter components are qualified by manufacturer (Operating Instruction Manual OI/MT 5000 – EN, Revision H 05.2013) (ABB) to operate between -40 deg F to 170 deg F, per Westinghouse letter LTR-SFPIS-16-07 Revision 1. As a result, Oyster Creek installed location for transmitter is bounded by manufacturer qualified range for operation.

Humidity: Per SDD-OC-811-A DIV1, Rev 1, Fire Protection New Cable Spreading Room & Cable Bridge Tunnel document, the cable bridge tunnel area, where the transmitters will be installed, can experience humidity between 0 – 100%. Transmitters are qualified by Westinghouse to operate between 0-95% humidity per EQ-QR-269. Transmitter housings are NEMA-4X, which means they are leak tight. There are two connection points for the transmitter. One for coaxial cable and one for 4 - 20mA cable connection point. Coax cable connector to transmitter housing is qualified by Westinghouse to be leak tight. This is the same straight connector that Westinghouse qualified to install in SFP area to connect coax cable to the probe. For straight connector Westinghouse recommends to install Raychem boot seal to achieve water tight seal (WNA-TR-3149). Oyster Creek installed Raychem boot seal for the straight connector since the transmitter is installed in a location that can experience 100% humidity condition to achieve water tight connection. As for 4 - 20mA connection point, Oyster Creek installed water tight seal to prevent condensation intrusion into the connection point.

Display Enclosures:

Radiation: Oyster Creek Environmental Qualification Report ES-027 Revision 4 states radiation limit "Not Available" where the display enclosures are installed. The area is a non-harsh environment. Per the radiological survey map for this location during normal operation, the radiation in this area (Upper Cable

Spreading Room) will be <1 mR/hr, and as a result is not expected to exceed 1.E03 Rads.

Temperature: Per 15050-M4-001 "Calculated Temperature in Upper Cable Spreading Room without Ventilation", the upper cable spreading room area, where the display enclosures will be installed, can experience temperatures between 40 - 102 deg F. For BDB conditions, display enclosures are qualified up to 140 deg F. As a result, there is no concern with operating display enclosures in upper cable spreading room area during BDB event conditions. During normal operating conditions, display enclosures are qualified by Westinghouse to operate between 50 deg F – 120 deg F per EQ-QR-269. However, upon further investigation, Westinghouse found components within display enclosures are qualified by the manufacturers to operate between -4 deg F to 158 deg F, per Westinghouse letter LTR-SFPIS-16-07 Revision 1. As a result, Oyster Creek installed location for display enclosures is bounded by manufacturer qualified range for operation.

Humidity: Per SDD-OC-811-A DIV1, Rev 1, Fire Protection New Cable Spreading Room & Cable Bridge Tunnel document, the upper cable spreading area, where the display enclosures will be installed, can experience humidity between 0 – 100%. Display enclosure are qualified by Westinghouse to operate between 0 – 95% humidity per EQ-QR-269. Though the SDD-OC-811-A DIV1, Rev 1, Fire Protection New Cable Spreading Room & Cable Bridge Tunnel document states humidity can be up to 100% in upper cable spreading area, the area is in a building that does not have any sources of water that can lead up to 100% humidity. Also, the display enclosures are a NEMA-4X enclosure per Westinghouse Report WNA-TR-03149. There are two connection points to each display enclosure. One for 4-20 mA cable connection to display enclosure and one for power conduit connection. To enhance these two connections to make them water intrusion proof when operating in 100% humidity condition, Oyster Creek added water tight seals to both connection points to display enclosure.

Since the transmitter and display enclosure are installed in a location that is outside Westinghouse qualified range for operation (even though manufacturer individually qualified the components beyond Westinghouse qualified range), Westinghouse recommends performing additional monitoring (once per shift) of the level display when operating outside the qualified environment (50 - 120 deg F), per Westinghouse letter LTR-SFPIS-16-07 Revision 1. Oyster Creek will perform these additional monitoring steps to ensure equipment is functioning properly when operating outside the qualified operating range. Oyster Creek will obtain display level form both primary and backup channels and will compare it to actual water level in SFP by performing a physical walkdown. This is to ensure both displays are functioning properly against the real water level in the SFP. The acceptance criteria for declaring functionality will be ± 1 ft. when operating outside qualified range for operation. This acceptance criterion is in accordance with the NEI 12-02 Revision 1 guidance requirements.

In addition to enhanced monitoring, Oyster Creek will also reduce time to implement compensatory measures when one channel is found non-functional from 90 days to 45 days, when operating outside the qualified range. This will ensure instruments are returned to a functional status in a timely manner.

During the onsite audit, the NRC staff reviewed Calculation ES-027, Revision 4, "Environmental Parameters Oyster Creek NGS; SDD-811A, Revision 1, Fire Protection – New Cable Spreading Room and Cable Bridge Tunnel Oyster Creek Nuclear Generating Station"; Calculation 15050-M4-001 Revision 8, "Calculation Temperature in Upper Cable Spreading Room without any Ventilation"; and Westinghouse letter LTR-SFPIS-16-07, Rev, 1, "SFPIS: Engineering Judgement for Exelon – Oyster Creek". The staff confirmed that environmental conditions for SFPLI in the SFP area are bounded by those in Westinghouse environmental test report EQ-QR-269. The radiation and temperature for which the transmitters and display enclosures are qualified, envelop the normal and BDBEE conditions where the equipment is located. For humidity environmental qualification, the transmitter and display enclosure housing are NEMA-4X which are qualified for 100 percent humidity. For connection points to the transmitter and display enclosure, the licensee will install water tight seals to prevent condensation intrusion into connection points. In addition, the licensee will monitor the display once per shift when operating outside the temperature qualified range as well as reducing the out of service time (OOS) from 90 days to 45 days. The combined compensatory measure and shorter OOS time would ensure that the transmitter and display enclosure are functioning properly when operating outside the temperature qualified testing range.

4.2.4.2.2 Electromagnetic Compatibility

During the onsite audit, the NRC staff inquired about an assessment of potential susceptibilities of Electromagnetic Interference (EMI) and Radio-Frequency Interference (RFI) in the areas where the SFP instruments are located and how to mitigate those susceptibilities. The licensee responded that potential susceptibilities of EMI/RFI in the area where the SFP instrument components, (i.e. probe/coupler connections, transmitters, display enclosures) are located will be evaluated and identified per Modification Acceptance Test Plan under WO C2035047. Acceptance test criteria will determine whether a radio-free zone is needed. The staff later reviewed WO C2035047, Post Modification Acceptance Test (PMAT) for Primary and Secondary SFPLI. In this PMAT, the licensee operated a station radio near spent fuel pool probe, SFPLI transmitter and SFPL electronics panel. The licensee also operated the SFP bridge crane near to the SFPLI equipment. There was no disturbance seen on the level indicator and no radio zone was identified. The staff found the licensee response acceptable because the licensee has performed a PMAT and found no EMI/RFI interference near the SFP instrument components.

Based on the evaluation above, the NRC staff finds the licensee's proposed instrument qualification process appears to be with consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately addresses the requirements of the order.

4.2.5 Design Features: Independence

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

a) The level probes will be mounted on the south side of the SFP and will be separated by a distance greater than the span of the shortest side of the pool. This meets the NEI 12-02, Revision 1 guidance for channel separation.

b) The coax cables from the level instruments in the SFP area are routed directly one floor below into the reactor building elevation 95-ft. 3-in. and as a result the separation between the coax cables in the refuel floor area is maintained by a distance greater than the span of the shortest side of the pool. Outside of the refuel floor area, the coax cables maintain a minimum separation of 8ft. The primary coaxial cable will be installed in conduit and the secondary coaxial cable will be installed in combination of conduit and cable tray. The primary level transmitter is installed on the south Cable Tray Bridge and the secondary level transmitter is installed in the north Cable Tray Bridge on elevation 80 ft. The two level transmitters are physically separated of approximately 15 ft. The indicator electronics / UPS enclosures for the primary and secondary instrument channels are installed in a non-hazardous area, the Upper Cable Spreading Room elevation 68 ft.-6 in. with a physical separation of 5 ft. The 4-20mA cables that extend from the transmitters to the displays have a minimum separation of 1 in., maintaining the 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. The primary 4-20mA cable will be installed in conduit and the secondary 4-20mA cable will be installed in combination of conduit and cable tray. Distribution panels PDP-733-57 & PDP-733-58 are located in the Lower Cable Spreading Room. The 120VAC cables from the distribution panels terminate to panel 17R in the Control Room and to the displays in the Upper Cable Spreading Room. The 120VAC power cables are 1 in. apart, maintaining the 1E separation requirements per Oyster Creek Station Specification SP-9000-41-005, Cables & Raceways. All the equipment and supports will be seismically mounted. The 120VAC distribution panels for the primary and secondary instruments are powered by different 480V safety buses. Therefore, the loss of any one bus will not result in the loss of ac power to both instrument channels.

During the onsite audit, the staff performed a walkdown to the SFPLI channels. The staff found that the primary instrument channel is independent of the backup instrument channel in accordance with recommendations in NEI 12-02 for channel independence.

Based on the evaluation above, the NRC staff finds that the licensee's proposed design, with respect to instrument channel independence, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.6 Design Features: Power Supplies

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

a) The primary SFPLI instrument channel will be normally powered from safety related 120 VAC, Post Accident Instrument Panel PDP-733-58. The backup SFPLI instrument channel will be normally powered from the safety related 120 VAC, Post Accident Instrument Panel PDP-733-57. The panels are supplied by different safety buses, which maintain power source independence, from 4160V buses to the instrument channels. The channel powered by Panel PDP-733-58 is powered from 4160V bus D and the channel powered by Panel PDP-733-57 is powered from 4160V bus C. Upon loss of normal ac power, individual batteries installed in each channel's electronics / UPS enclosure will automatically maintain continuous channel operation for at least 3 days. Before reaching the 3-day battery life, the Post Accident Instrument Panels will be restored using the FLEX diesel generator. Spatial and physical separation of the power cables will be maintained and comply with Oyster Creek Station Specification SP-9000-41-005, Cable & Raceways.

b) Westinghouse Report, WNA-CN-00300-GEN, provides the results of the calculation depicting the battery backup duty cycle. This calculation demonstrates that battery capacity is 4.22 days to maintain the level indicating function to the display location. The displays are located in the Upper Cable Spreading Room elevation 63'-6". The results of the calculation meet the NEI-12-02 requirements.

During the onsite audit, the NRC staff reviewed Drawings GU 3D-644-16-001 Revision 3, "Post Accident Monitoring Block Diagram 120 VAC Vital PWR to PNL 2R, 3R, 5R, 5F/6F, 16R &17R; and BR 3000 Revision 14, "Electrical Power System Key One Line Diagram". The staff found that the electrical ac power sources for the primary and backup channels are independent and the loss of power supply from one channel will not result in a loss of ac power for both channels. The staff reviewed the battery duty cycle during the vendor audit and found it to be acceptable. Based on the evaluation above, the NRC staff finds that the licensee's proposed power supply design appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.7 Design Features: Accuracy

In a letter dated August 26, 2016 [Reference 33], the licensee stated that:

a) The Westinghouse documents WNA-CN-00301 and WNA-DS-02957-GEN describe the channel accuracy under both (a) normal SFP level condition and (b) at the Beyond Design Basis (BDB) condition that would be present if SFP level were at Level 2 and Level 3 datum points. Each instrument channel will be accurate to within $\pm$ 3" during normal spent fuel pool level conditions. The instrument channels will retain this accuracy after BDB conditions, in accordance with the above Westinghouse documents. This value is within the channel accuracy requirements of the Order (+/- 1 ft.).

b) The Westinghouse document WNA-TP-04709-GEN calibration and technical manual describes the methodology for routine testing/calibration verification and calibration methodology. This document also specifies the required accuracy criteria under normal operating conditions. Oyster Creek Station will follow the guidance and criteria provided in this document. Instrument channel calibration will be performed if the level indication reflects a value that is outside the acceptance band provided in Westinghouse documents. Calibration will be performed once per refueling cycle for Oyster Creek Station. Per Westinghouse document WNA-TP-04709-GEN, calibration on a SFP level channel is to be completed within 60 days of a planned refueling outage considering normal testing scheduling allowances (e.g., 25%). This is in compliance with the NEI 12-02 guidance for Spent Fuel Pool Instrumentation.

In accordance to the licensee's letter, the staff found the channel accuracy under normal and BDBEEs are consistent with Westinghouse's specified accuracy. This accuracy is within 1 ft. accuracy identified in NEI 12-02. This accuracy will be used as an acceptance criterion for a calibration procedure to flag to operators and to technicians that the channel requires adjustment to within the normal condition design accuracy.

Based on the evaluation above, the NRC staff finds that the licensee's proposed instrument accuracy appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.8 Design Features: Testing

In a letter dated August 26, 2016 [Reference 33], the licensee stated that:

a) Westinghouse provided test equipment that provides the capability to calibrate the equipment. Westinghouse calibration procedure WNA-TP-04709-GEN provide instructions to use the test equipment to perform a calibration. The Westinghouse calibration procedure is included in the Oyster Creek's site vendor manual VM-OC-6652. Oyster Creek will perform an in-situ test during the functional check based on the Westinghouse Two Point Verification Method, LTR-SFPIS-14-55.

b) The levels displayed by the channels and other SFP level instrumentation shall be checked against each other and checked against physical pool level per the Oyster Creek Station processes. If the level is not within the required accuracy per Westinghouse recommended tolerances, channel calibration will be performed.

c) Oyster Creek will perform a PM [preventive maintenance] to perform a functional check 60 days prior to a refuel outage. The PM establishes the current water level by measuring the distance to the water referenced from the bottom of the launch plate. This measured distance is then compared to the level indication obtaining the As-Found indication value. The probe is then lifted out of the water to a predetermined mark on the probe. The water level at the predetermined mark is then recorded. The probe is lowered back into the water freely suspended from the launch plate. The level indication is recorded. If the As-Found Level indication is within the tolerance specified, the procedure is exited. If the As-Found value is not within tolerance the calibration is performed. The PM and calibration steps were taken from Westinghouse document WNA-TP-04709-GEN Spent Fuel Pool Instrument System Calibration Procedure and Westinghouse Two Point Verification Method, LTR-SFPIS-14-55.

d) Oyster Creek has developed a preventive maintenance task to perform a functional check 60 days prior to a refuel outage. Oyster Creek has developed a preventive maintenance task to perform an annual visual inspection of the sensor probe (full range) to ensure there is no damage or corrosion.

During the onsite audit, the NRC staff reviewed WNA-TP-04613-GEN Revision 5, "Spent Fuel Pool Instrumentation System Functionality Test Procedure" and WNA-TP-04709-GEN, Revision 4, "Spent Fuel Pool Instrumentation System Calibration Procedure" and found that these procedures adequately described periodic testing, calibration, channel check, function check, and preventive maintenance to ensure the SFPLI channels are reliable to perform their intended function. These actions are consistent with those recommended by Westinghouse.

Based on the evaluation above, the NRC staff finds that the licensee's proposed SFP instrumentation design allows for testing consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.9 Design Features: Display

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

> a) The Oyster Creek Station primary and backup instrument channel displays are located in the Upper Cable Spreading Room which is a robust seismic category I structure.
>
> b) The Oyster Creek Station primary and backup instrument channel displays are located in the Turbine Building on elevation 63'-6", the floor above the Control Room. The Upper Cable Spreading Room is accessible through a staircase adjacent to the Control Room. The Upper Cable Spreading Room, Control Room and the path between the two locations are considered non-harsh environments per Engineering Standard ES-027, "Environmental Parameters Oyster Creek NGS"; therefore, the radiological exposure at these locations shall remain less than emergency exposure limits allowable for emergency responders to perform this action. Heat and humidity from SFP boil-down conditions are not applicable in these locations. The locations are several floors below the SFP operating floor and located in a different building physically separated by concrete walls, such that heat and humidity from a boiling SFP would not compromise habitability at these locations.
>
> During ELAP conditions radiation due to boiling of SFP is not an issue for Oyster Creek because SFP is located in a different building compared to where the displays are located (SFP is in reactor building and displays are in Turbine Building). Also, during ELAP there is no LOCA assumed simultaneously. During normal operating conditions the dose in the traverse path and display location is nominal (< 1mRem/hr.) which should not change as LOCA is not assumed during ELAP.
>
> The temperature in the traverse path and in the upper cable spreading room (where the display enclosures will be installed) will be maximum 99 deg F without ventilation, per Calc. 15050-M4-001, Revision 2.

> The humidity in the traverse path and in upper cable spreading room can be up to 100% per ES-027, Revision 4. The path to display enclosures from MCR is very short (2 minutes or less) and operators should take very minimal time to read out the display as it's readily available. As a result, humidity should not be a concern for operators to traverse through this path.
>
> Spent Fuel Pool Level monitoring will be the responsibility of Operations personnel who will monitor the display periodically once dispatched from the Control Room. Travel time from the Control Room to the primary and secondary displays is approximately 2 minutes based on walkdowns performed. Diverse communications are accessible at both display locations. The operators would employ radio communications or the telephone communication. If the radio communications or telephone systems are nonfunctional, the sound powered phone system shall be available. A sound powered phone jack is located in the Control Room; however, a sound powered phone jack is not located in the Upper Cable Spreading Room. The sound powered phone will be available to dispatch personnel for the monitoring of the indicators. The operator can then physically report back the level to the Control Room. The display will be accessed on demand.

During the onsite audit, the NRC staff walked down to the display locations. The staff also reviewed Calculation 15050-M4-001 Revision 2, "Upper Cable Spreading Room Temperature without Any Ventilation" and DDQ -16-00454, "OCNGS Radiological Survey." The staff found that the display location is promptly accessible and habitable during a BDBEE.

Based on the evaluation above, the NRC staff finds that the licensee's proposed location and design of the SFP instrumentation displays appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3 Evaluation of Programmatic Controls

Order EA-12-051 specified that the spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation programmatic controls, including training, procedures, and testing and calibration. Below is the NRC staff's assessment of the programmatic controls for the spent fuel pool instrumentation.

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that:

> The Systematic Approach to Training (SAT) will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training. Training will be completed prior to placing the instrumentation in service.

Guidance document NEI 12-02 specifies that the SAT process can be used to identify the population to be trained, and also to determine both the initial and continuing elements of the required training.

Based on the OIP statement above, the NRC staff finds that the licensee's plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFP level instrumentation, including the approach to identify the population to be trained, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.2 Programmatic Controls: Procedures

In its letter dated August 26, 2016 [Reference 33], the licensee stated that:

> Performance tests (functional checks) and Operator performance checks will be described in detail in the vendor operator's manual, and the applicable information is planned to be contained in plant operating procedures. Operator performance tests will be performed periodically as recommended by the equipment vendor. Channel functional tests per Operations Procedures, with limits established in consideration of vendor equipment specifications, will be performed at appropriate frequencies established in the SFPIS manual. Operator performance checks will be performed on a periodic scheduled basis, with additional maintenance on an as-needed basis when flagged by the system's automated diagnostic testing features. Channel calibration tests per maintenance procedures, with limits established in consideration of vendor equipment specifications, will be performed at frequencies established in consideration of vendor recommendations. SFPIS channel/equipment maintenance, preventative maintenance, and testing program requirements to ensure design and system readiness will be established in accordance with Exelon's processes and procedures and in consideration of vendor recommendations to ensure that appropriate regular testing, channel checks, functional tests, periodic calibration, and maintenance is performed (and available for inspection and audit). Subject maintenance and testing program requirements will be developed during the SFPIS modification design process.
>
> The Westinghouse calibration procedure and technical manual are embedded into the Oyster Creek SFPIS vendor manual VM-OC-6652. Oyster Creek has developed a program document CC-OC-118, "Oyster Creek Implementation of Diverse and Flexible Coping Strategies (FLEX) and Spent Fuel Pool Instrumentation Program" defining the requirements of the SFPIS.
>
> Oyster Creek developed Operator Round guidance to perform channel checks against each other and checks against physical pool level.
>
> The program document CC-OC-118, "Oyster Creek Implementation of Diverse and Flexible Coping Strategies (FLEX) and Spent Fuel Pool Instrumentation Program" provides guidance for taking compensatory actions when either one or both channels are out of service. This guidance will be incorporated into implementing operating procedures.

During the onsite audit, the licensee provided a sample of implementing procedures. The staff reviewed calibration procedure WNA-TP-04709-GEN, Revision 4, "Spent Fuel Pool Instrumentation System Calibration Procedure" and found it acceptable. The licensee will develop the list of procedures addressing operation, calibration, test, maintenance, and inspection procedures for the SFPLI. According to the licensee, these procedures will be consistent with Westinghouse's recommendations.

Based on the evaluation above, the NRC staff finds that the licensee's procedure development appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

In its letter dated August 26, 2016 [Reference 33], in addition to the sections quoted directly above, the licensee also stated that:

> Both primary and backup SFPIS channels incorporate permanent installation (with no reliance on portable, post-event installation) of relatively simple and robust augmented quality equipment. Permanent installation coupled with stocking of adequate spare parts reasonably diminishes the likelihood that a single channel (and greatly diminishes the likelihood that both channels) is (are) out-of-service for an extended period of time. Planned compensatory actions for unlikely extended out-of-service events will be controlled by Procedure CC-OC-118, Diverse and Flexible Coping Strategies (Flex) and Spent Fuel Pool Instrumentation Program Implementation, and are summarized as follows:

The licensee stated that for 1 channel out of service the required restoration action is as follows:

> Restore channel to functional status with 90 days (or within 45 days when operating outside qualified range), then proceed to Compensatory Action.

The licensee stated that for 2 channels out of service the required restoration action is as follows:

> Initiate action within 24 hours to restore one channel to functional status and restore one channel to functional status within 72 hours.

The licensee stated that compensatory action is as follows:

> Initiate an Issue Report to enter the condition into the Corrective Action Program. Identify the equipment out of service time is greater than the specified allowed out of service time, develop and implement an alternate method of monitoring, determine the cause of the non-functionality , and the plans and schedule for restoring the instrumentation channel(s) to functional status.

The NRC staff finds that the licensee has adequately described maintenance, testing, channel checks, and functional tests. These maintenances and tests are consistent with Westinghouse recommendations. The staff also finds that the out-of-service allowed outage time and compensatory measure that will be taken are consistent with those recommended in NEI 12-02 and therefore are acceptable.

Guidance document NEI 12-02 contains provisions for the establishment of processes that will maintain the SFPLI at their design accuracy. It also contains provisions for the control of surveillance and allowed-outage-time for each channel. Based on the licensee's letter referenced above, the NRC staff finds that the licensee's proposed testing and calibration processes are consistent with vendor recommendations and the provisions of NEI 12-02. Further, the licensee's proposed restoration actions and compensatory measures for the instrument channel(s) out-of-service are consistent with NEI 12-02.

Based on the evaluation above, the NRC staff finds that the licensee's proposed testing and calibration plan appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.4 Conclusions for Order EA-12-051

In its letter dated December 1, 2016 [Reference 35], the licensee stated that they would meet the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee has conformed to the guidance in NEI 12-02, as endorsed by JLD-ISG-2012-03. In addition, the NRC staff concludes that if the SFP level instrumentation is installed at Oyster Creek according to the licensee's proposed design, it should adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit in August 2016 [Reference 17]. The licensee reached its final compliance date on December 6, 2016 [Reference 18], and has declared that OCNGS is in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs that if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders.

6.0 REFERENCES

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OYSTER CREEK NUCLEAR GENERATING STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051
DATED APRIL 19, 2017

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