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

December 14, 2016

Mr. William R. Gideon
Site Vice President
Brunswick Steam Electric Plant
8470 River Rd. (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – 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. MF0975, MF0976, MF0973, AND MF0974)

Dear Mr. Gideon:

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. ML13071A559), Duke Energy Progress, LLC (Duke, the licensee) submitted its OIP for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, 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 November 22, 2013 (ADAMS Accession No. ML13220A090), and March 31, 2015 (ADAMS Accession No. ML15082A155), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated June 3, 2015 (ADAMS Accession No. ML15173A013), Duke reported that BSEP, Unit 2 was in full compliance with Order EA-12-049. By letter dated May 19, 2016 (ADAMS Accession No. ML16146A604), Duke reported that BSEP, Unit 1 was in full compliance with Order EA-12-049 and submitted a Final Integrated Plan for BSEP, Units 1 and 2.

By letter dated February 28, 2013 (ADAMS Accession No. ML13086A096), Duke submitted its OIP for BSEP 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 18, 2013 (ADAMS Accession No. ML13269A345), and March 31, 2015 (ADAMS Accession No. ML15082A155), 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 June 3, 2015 (ADAMS Accession No. ML15173A013), Duke reported that BSEP, Unit 2 was in full compliance with Order EA-12-051. By letter dated May 19, 2016 (ADAMS Accession No. ML16146A604), Duke reported that BSEP, Unit 1 was in full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Duke's strategies for BSEP, Units 1 and 2. The intent of the safety evaluation is to inform Duke 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 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML14273A444). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact Peter Bamford, Orders Management Branch, BSEP Project Manager, at 301-415-2833 or at Peter.Bamford@nrc.gov.

Sincerely,

Mandy K Halter

Mandy K. Halter, Acting Chief
Orders Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket Nos.: 50-325 and 50-324

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

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 (SFPLI) 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 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 alternating current (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 August 21, 2012, following several submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, "Diverse and Flexible

Coping Strategies (FLEX) Implementation Guide," Revision 0 [Reference 6] to the NRC to provide 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 and on August 29, 2012, issued its final version of Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, "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 0, with comments, as an acceptable means of meeting the requirements of Order EA-12-049, and published a notice of its availability in the *Federal Register* (77 FR 55230).

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 SFPLI. 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], Duke Energy Progress, LLC (Duke, the licensee) submitted an Overall Integrated Plan (OIP) for Brunswick Steam Electric Plant (BSEP, Brunswick), Units 1 and 2, in response to Order EA-12-049. By letters dated August 20, 2013 [Reference 11], February 28, 2014 [Reference 12], August 28, 2014 [Reference 13], February 27, 2015 [Reference 14], August 26, 2015 [Reference 15], and February 24, 2016 [Reference 16], the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 17], 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 November 22, 2013 [Reference 18], and March 31, 2015 [Reference 19], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated June 3, 2015 [Reference 20], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved for BSEP, Unit 2. By letter dated May 19, 2016 [Reference 21], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved for BSEP, Unit 1, 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.

Brunswick, Units 1 and 2, are General Electric Model 4 boiling-water reactors (BWRs) 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.

At the onset of an ELAP both reactors are assumed to trip from full power. The main condenser is assumed to be unavailable due to the loss of circulating water. Decay heat is removed when the safety relief valves (SRVs) open on high pressure and dump steam from the reactor pressure vessel (RPV) to the suppression pool located in the containment. Makeup to the RPV is provided by the reactor core isolation cooling (RCIC) turbine-driven pump. The normal alignment for the RCIC pump is to draw suction from the condensate storage tank (CST). For BSEP, the CSTs have been evaluated to be robust structures. Within 1 hour, BSEP will swap suction from the CST to the suppression pool to conserve CST inventory. When suppression pool water temperature reaches 190 degrees Fahrenheit (°F), BSEP will swap RCIC pump suction back to the CST. Using this approach, replenishment of the CST inventory will not be required for approximately 52 hours. Maintaining RCIC suction temperature below 190°F ensures sufficient net positive suction head (NPSH) for the RCIC pump. Within 1 hour, the operators take manual control of the SRVs to perform a controlled cooldown and depressurization of the reactor. The cooldown of the primary system is stopped when reactor pressure reaches a control band of 150 pounds per square inch gauge (psig) to 300 psig to ensure sufficient steam pressure to operate the RCIC pump. The RPV makeup will continue to be provided from the RCIC system until the gradual reduction in RPV pressure resulting from diminishing decay heat requires a transition to Phase 2 methods. The RCIC injection source will be maintained for as long as possible, since it is a closed loop system using relatively clean CST and suppression pool water.

When the RCIC system is no longer available, the preferred RPV makeup supply in Phase 2 comes from a portable FLEX pump. The portable FLEX pump will be aligned to existing systems using pre-designated hose and FLEX connections to inject water into the RPV. The credited sources of water for the FLEX pumps include the CST and the discharge canal. Other sources of water that may be available, but are not credited as part of the FLEX strategy, include the demineralized water tank, the fire water tank, and condensate hotwells. Brunswick will use sources with higher quality water first. Raw water will be used only if necessary.

Both reactors have Mark I containments which are inerted with nitrogen at power. Brunswick has developed their FLEX response strategies to not rely on "anticipatory venting" of containment. If required, BSEP would vent containment through the hardened wetwell vent prior to reaching the primary containment pressure limit (PCPL) of 70 psig. The hardened wetwell vent can be powered by the FLEX diesel generators (DGs), which would be operating prior to containment reaching the PCPL. Brunswick would continue venting through the hardened wetwell vent, as necessary.

Each BSEP unit has a SFP in its Reactor Building. To maintain SFP cooling capabilities, the licensee stated that the required action is to establish the water injection lineup before the environment on the SFP operating deck degrades due to boiling in the pool so that personnel can access the refuel floor to accomplish the coping strategies. 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 5 hours after the start of the ELAP. Makeup to the SFP will not be required until approximately 57 hours into the event, which provides ample time to deploy the Phase 2 SFP makeup strategy. Brunswick will open the Reactor Building roof hatch within approximately 2 hours of the initiating event to support Reactor Building cooling and provide a vent path for SFP boil-off.

To makeup to the SFP, the licensee has three options. The primary SFP makeup strategy is to connect hoses from the FLEX pump discharge to a FLEX connection on the residual heat removal (RHR) fuel pool cooling assist line. An alternate strategy is to connect a hose from the discharge of the FLEX pump and route it directly to the SFP. The third option is to route hose from the FLEX pump discharge to two spray nozzles so that water can be sprayed over the spent fuel. For the latter two options, spray hoses and/or nozzles will be deployed early in the event.

Using electric power from the station batteries and pneumatic supplies from the nitrogen backup systems, the SRVs will remain functional following an ELAP. In addition to powering the SRVs, the station 125/250 Volts-dc (Vdc) Division II batteries will also power the RCIC system and vital instrumentation. Brunswick has permanently pre-staged two 480 volts-ac (Vac) 500 kilowatt (kW) FLEX DGs that can provide power within one hour of event initiation. Associated cabling is also permanently pre-staged so deployment consists only of racking in a 480 Vac breaker at a plant emergency bus and starting the FLEX DGs. The permanently pre-staged FLEX DGs will supply power to emergency busses associated with Division II on each unit. Either FLEX DG can be aligned to both emergency busses, energizing both units' Division II battery chargers. Because only Division II is credited for FLEX strategies, the permanently pre-staged FLEX DGs are fully redundant. In practice, both permanently pre-staged FLEX DGs would be placed into service and the capacity of both FLEX DGs will enable use of Division I battery chargers and Control Building heating, ventilation, and air conditioning (HVAC) equipment. If these FLEX DGs are not immediately available, BSEP can perform battery load shedding, which will extend availability of dc power from the batteries to 2 hours and 10 minutes.

In addition, during Phase 3, a National Strategic Alliance of FLEX Emergency Response (SAFER) Response Center (NSRC) will provide additional equipment (e.g., pumps, DGs) to back up the on-site FLEX equipment. The pumps supplied by the NSRC are compatible with the FLEX connections used at BSEP. The NSRC generators will be connected to the electrical distribution system using BSEP-specific cables and connectors that are stored in the BSEP FLEX Storage Building (FSB).

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 summary of compliance elements section of the licensee's compliance letters for each unit [References 20 and 21] indicate that the BSEP program document, procedures, and training have all been developed in accordance with NEI 12-06, Revision 0. Thus, the NRC staff evaluated the licensee's strategies against the endorsed NEI 12-06, Revision 0 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, both units 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 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 event are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions where one or more units are 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

According to the licensee's FIP, the injection of cooling water into the RPV will be accomplished through the RCIC system. Because the turbine for the RCIC pump is driven by steam from the RPV, operation of the RCIC system further assists the SRVs with RPV pressure control. The RCIC system suction is initially lined up to the CST and will pump water into the core from the CST. Both the CST and the RCIC pump are protected from all applicable hazards. In order to maximize the available volume of water in the CST the suction of the RCIC pump will be swapped to the suppression pool within one hour of the ELAP event. The RCIC pump will maintain suction from the suppression pool until a suppression pool temperature of 190°F is reached. Due to RCIC NPSH concerns, the suction will then be swapped back to the CST's. This strategy will extend the time until CST makeup is necessary to 52 hours.

Per the FIP, pressure control of the RPV is accomplished using the SRVs which are powered from the 125 Vdc buses. At approximately 1 hour after the initiation of the event, operators will utilize the SRV's to depressurize the RPV to 150-300 psig at a cooldown rate of less than 100°F per hour. After this point, the RPV pressure is maintained between 150 and 300 psig to allow for continued operation of the RCIC system. There is a backup nitrogen system in place to allow the continued operation of the SRV's for at least 24 hours after the initiation of the ELAP event.

3.2.1.2 Phase 2

In the BSEP FIP, the licensee states that RCIC will continue to be used until it is necessary to transfer to a portable FLEX pump. To support this operation, two permanently pre-staged and electrically connected 500 kW, 480 Vac FLEX DGs will be utilized to repower two load centers. Both of these DGs are fully redundant. These load centers will power the necessary battery chargers to ensure that dc powered components in the RCIC system, the SRV's, and critical instrumentation will continue to have power.

The use of these permanently pre-staged FLEX DGs is considered an alternative to NEI 12-06 and is further discussed in Section 3.14 of this safety evaluation (SE).

According to the licensee's FIP, prior to depletion of the backup nitrogen system, which supports control of the SRVs, a FLEX air compressor will be brought to a location between the Reactor Building and the Turbine Building. The air compressor will be connected to either a primary or alternate connection point. The primary strategy is to use a connection at a nitrogen backup remote supply line isolation valve. This connection is robust for all hazards except wind generated missiles. The alternate strategy is to use a connection point on the non-interruptible instrument air system inside the Reactor Building. This connection is robust for all hazards except seismic.

When RCIC is no longer available, the makeup strategy transitions to low pressure RPV makeup. The preferred suction strategy involves the portable FLEX pump using water from the CST. A discharge hose connected to the FLEX pump connects to a penetration in the Reactor Building wall. A hose is connected to the other side of the penetration and directs flow to the reactor water clean-up system and subsequently the RPV. The alternate makeup flow path would discharge water from the FLEX pump through a discharge hose connected to the integrated leak rate testing system (ILRT) piping, which can be connected to the RHR system to supply water to the RPV.

Alternatively, a FLEX pump can be used to provide water from the UHS for RPV makeup. If all pumps work adequately, the Extreme Damage Mitigation Guideline (EDMG) pump will be utilized to supply water to both plants CSTs from the plant discharge canal, and the FLEX pumps can remain connected as previously described. In the case that one of the two FLEX pumps fails, both of the remaining pumps (FLEX pump and EDMG pump) can take suction from the discharge canal and supply water to the primary or alternate connection points.

According to the licensee's FIP, the usage of the discharge canal will provide adequate volume for the core cooling strategy if needed in Phase 2 and into Phase 3. Each of the two diesel driven FLEX pumps is rated for 365 gallons per minute (gpm) at 285 per square inch absolute (psia). The EDMG pump is rated at 310 gpm at 269 psia. Brunswick performed a hydraulic analysis to confirm that the FLEX pumps and EDMG pump are adequately sized. Specifically, these calculations demonstrated that both the FLEX pumps and the EDMG pump can provide the required flow rates necessary to satisfy the requirements for both the core cooling and the SFP spray strategies.

In the event that raw water is used to provide core cooling, the licensee's FIP states that BSEP will utilize the guidance contained in BWR Owners Group (BWROG) report BWROG-TP-14-006, "Fukushima Response Committee Raw Water Issue: Fuel Inlet Blockage from Debris." This guidance contains direction to maintain the RPV water level at the level of the moisture separator drains to ensure that core cooling is maintained despite the possible clogging of fuel element orifices and filters. The licensee's FIP also states that BSEP would restore on-site capabilities for providing clean water makeup, or arrange for an off-site water source, should water from the discharge canal be utilized.

3.2.1.3 Phase 3

According to the licensee's FIP, Phase 3 strategies are the same as those for Phase 2. The licensee states that Phase 3 NSRC equipment will act as a backup or redundant equipment for Phase 2 strategies.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

In its FIP, the licensee stated that flooding events will not impact the deployment of FLEX equipment from the FSB to the planned locations in the FLEX strategy because the postulated floods that could reach the plant grade do not have sufficient persistence to impact the overall timeline. On-site flooding is discussed in Section 3.5.2 of this SE.

3.2.3 Staff Evaluations

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

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

Phase 1

The licensee's Phase 1 core cooling FLEX strategy relies on the existing RCIC system to maintain water level in the RPV. As described in the FIP, Sections 2.3.4.2 and 2.3.4.3, the RCIC system is robust and located in the Reactor Building which is protected from all applicable hazards. The licensee stated in its FIP, that the RCIC system was originally designed and constructed to be seismically qualified, but had not been maintained as such. However, the licensee performed a new evaluation confirming that RCIC, as used for the mitigating strategies plan, is seismically robust. As described in the FIP, RCIC will initially take suction directly from the CST, but will be realigned to the suppression pool to conserve CST inventory. The licensee will switch RCIC back to the CST when the suppression pool reaches 190°F to ensure the necessary NPSH is maintained. The RCIC control and instrumentation will be powered from station batteries or the permanently pre-staged FLEX DGs. Based on the licensee's FIP discussion, the NRC staff review concludes that the RCIC system is robust and should be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

Core decay heat will produce steam in the RPV, which will cause the SRVs to open on high steam pressure and relieve the steam to the suppression pool. Additionally, the licensee plans to eventually depressurize the RPV using the SRVs. As described in the BSEP FIP, the SRVs and associated nitrogen system are safety-related, and are located in the Reactor Building. As described in the FIP, the SRVs can be controlled using electric power from station batteries and a backup nitrogen supply for at least 24 hours during Phase 1, or they can actuate mechanically without power due to high pressure in the RPV. Based on the FIP discussion, the staff review concludes that the SRVs are robust and should be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

Phase 2

The licensee's Phase 2 core cooling strategy continues to use the suppression pool as the heat sink. Brunswick will continue to use the RCIC pump as long as possible, or transition to using a portable FLEX pump discharging through a primary or alternate connection point to the RPV. The licensee does not plan to rely on any installed plant SSCs other than the systems with FLEX connection points and certain water sources as discussed in Sections 3.7 and 3.10 of this SE, respectively.

Phase 3

The licensee's Phase 3 core cooling strategy initially relies on Phase 2 strategies with the NSRC equipment providing backup equipment. The licensee states in FIP Section 2.3.3 that the NSRC pumps are compatible with the Phase 2 connections.

3.2.3.1.2 Plant Instrumentation

The licensee's plan for BSEP is to monitor instrumentation in the control room and by alternate means to support the FLEX cooling strategy. The instrumentation is powered by station batteries and will be maintained for indefinite coping via battery chargers powered by the permanently pre-staged FLEX DGs.

As described in the BSEP FIP, the following instrumentation will be relied upon to support FLEX core cooling and inventory control strategy:

- RPV level (narrow range)
- RPV level (wide range)
- RPV level (fuel zone)
- RPV pressure (narrow range)
- RPV pressure (wide range)
- Containment (drywell/suppression pool) temperature.
- Suppression pool water level
- Drywell pressure
- CST level

These instruments are monitored from the control room and remote shutdown panel in the Reactor Building.

The NRC staff notes that the instrumentation identified by the licensee to support its core cooling strategy is consistent with the recommendation specified in the endorsed guidance of NEI 12-06.

Per the FIP, instrumentation is normally powered by station batteries. The batteries are being charged within 1 hour of the ELAP event by the permanently pre-staged FLEX DGs, which provide power to the battery chargers through the plant emergency buses. If the permanently pre-staged FLEX DGs are not immediately available, the battery power is extended by load shedding to maintain availability of instruments. The FLEX generators continue to provide power through all phases. Additional backup generators will be available from NSRC during Phase 3. Therefore, based upon the information provided by the licensee, the NRC staff concludes that the locations of the instrument indications should be accessible continuously throughout the ELAP event.

In accordance with NEI 12-06 Section 5.3.3.1, guidelines for obtaining critical parameters locally are provided in a FLEX Support Guideline (FSG). The BSEP guide 0EOP-01-FSG-08, "FLEX Instrumentation," provides alternate methods for obtaining critical parameters if key parameter instrumentation is unavailable. These critical parameters include RPV level and pressure, suppression pool level and temperature, containment temperature, and drywell pressure.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee's 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, the licensee has addressed both in a single, coupled calculation. This dependency notwithstanding, the NRC staff's discussion in this section of the SE solely focuses on the licensee's analysis of reactor core cooling. The review of the licensee's analysis of containment thermal-hydraulic behavior is provided in Section 3.4.4.2 of this evaluation.

The 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 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 simulations 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 [Reference 42], 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.

During the BSEP audit process, the NRC staff verified that the licensee's MAAP4 calculation, along with an associated addendum, addressed the limitations from the NRC staff's endorsement letter. The licensee utilized the generic roadmap and response template that had been developed by EPRI to support consistency in individual licensee's responses to the limitations from the endorsement letter. In particular, based upon review of the MAAP4 calculation documentation, the staff concluded that appropriate inputs and modeling options had been selected for the code parameters expected to have dominant influence for the ELAP event. The NRC staff further observed that the limitations imposed in the endorsement letter, particularly those concerning the RPV collapsed liquid level being maintained above the reactor core and the primary system cooldown rate being maintained within BSEP Technical Specification (TS) 3.4.9 limits, were satisfied. Specifically, the licensee's analysis calculated that BSEP would maintain the collapsed liquid level in the reactor vessel above the top of the active fuel region throughout the analyzed ELAP event. The licensee calculated that the minimum RPV water level above the top of active fuel is approximately 3 feet and occurs during the initial RPV depressurization. By maintaining the reactor core fully covered with water, adequate core cooling is assured for this event. Additionally, BSEP's fulfillment of the endorsement letter condition regarding the primary system cooldown rate signifies that thermally induced volumetric contraction and other changes in primary system thermal-hydraulic conditions should proceed relatively slowly with time, which supports the NRC staff's confidence in the predictions of the MAAP4 code. Furthermore, that the licensee should be capable of maintaining the entire reactor core submerged throughout the ELAP event is consistent with the staff's expectation that the licensee's flow capacity for primary makeup (i.e., installed RCIC pump and, subsequently, FLEX pumps) should be sufficient to support adequate heat removal from the reactor core during the analyzed ELAP event, including potential losses due to expected primary leakage.

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.

The licensee's calculations for BSEP assumed a total leakage rate at normal RPV operating pressure of 61 gpm. This leakage rate includes 18 gpm per recirculation pump seal in accordance with NRC Generic Letter (GL) 91-07 [Reference 44]. In addition, the licensee's calculation assumed a primary system leakage rate equal to the TS 3.4.4 limit of 25 gpm. Thus, between the two recirculation pumps and the additional primary system leakage, the total primary leakage rate assumed for BSEP during the ELAP event was 61 gpm at normal operating reactor pressure. This leakage rate was used in BSEP MAAP4 analysis.

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. In its FIP, the licensee stated that the seal leakage rate is a function of RPV pressure. Based on the licensee's analysis the limiting flow rate for the FLEX pump with respect to RPV makeup occurs at a discharge pressure of 285 psia. At this pressure the FLEX pump is able to provide 365 gpm at a time when core cooling required a minimum flow rate of 300 gpm. This required flow considers the flow required to replace evaporative losses as well as leakage losses. Further depressurization would result in a reduction in the leakage loss term.

Considering the above factors, the NRC staff concludes that the leakage rate of 18 gpm is reasonable based on the GL 91-07 content. Gross seal failures are not anticipated to occur during the postulated ELAP event. As is typical of the majority of U.S. BWRs, BSEP has an installed steam-driven pump (i.e., RCIC) capable of injecting into the primary system at a flow rate well in excess of the primary system leakage rate expected during an ELAP event, and the other pumps used for core cooling in its FLEX strategy have a similar functional capability. As discussed previously, at the limiting pressure the FLEX pump is able to inject at a rate which maintains adequate margin.

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 BSEP's Updated Final Safety Analysis Report (UFSAR) [Reference 44], Sections 4.2.1.1.8 and 4.3.1.5, the control rods provide adequate shutdown margin under all anticipated plant conditions, with the assumption that the highest-worth control rod remains fully withdrawn. Brunswick TS Section 1.1 (Definitions), further clarifies 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.

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

The licensee's FLEX strategy relies on one portable pump per unit during Phase 2. The FLEX pumps take suction from either the CST or the discharge canal and discharge to the RPV via installed connections. In Section 2.3.10 of its FIP, the licensee identified the performance criteria (e.g., flow rate, discharge pressure) for the Phase 2 portable pumps. The licensee's FIP also states that the Phase 3 medium flow pump specifications exceed the requirements identified by the BSEP hydraulic analysis for RPV and SFP makeup.

The licensee relies on one FLEX pump per unit to provide water for both long-term RPV and SFP makeup. The licensee procured two, diesel-driven FLEX pumps and credits the EDMG pump to satisfy the "N+1" (3 total) requirement. Section 2.3.10 of the FIP states that either FLEX pump can provide 2100 gpm at 150 psig. It further states that the EDMG pump can provide 1500 gpm at 150 psig. During the audit, the licensee provided calculation 0FLEX-0003, "Hydraulic Analysis for Fukushima FLEX Connection Modifications," Revision 0. The purpose of this calculation was to validate the flow qualification (discharge pressure, flow and NPSH) of the FLEX and EDMG pumps using Fathom modeling computer software. The hydraulic model takes into account both suction sources, the discharge canal and CST, and uses pump flow data from both pump types. Furthermore, the model conservatively assumes RPV injection of 300 gpm and SFP spray of 250 gpm (spray flow via portable nozzles).

The staff was able to confirm that flow rates and pressures evaluated in the hydraulic analyses were reflected in the FIP for the respective RPV and SFP makeup strategies. This conclusion is based upon the use of the described FLEX pumps and the 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 that the evaluations of the pump staging locations, hose distance runs, and connection points were consistent with the hydraulic analysis and FIP.

Based on the staff's review of the FLEX pumping capabilities at BSEP, as described in the above hydraulic analysis and the FIP, the NRC staff concludes that the portable FLEX pumps should perform as intended to support core and SFP cooling 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 loss of normal access to the UHS. 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, 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 HVAC caused by the event.

During the first phase of the ELAP event, BSEP 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 (core cooling, inventory control, and containment integrity). The BSEP Class 1E station batteries and associated electrical distribution systems are located within the Reactor Building, Control Building, and the Emergency Diesel Generator (EDG) Building, which are seismic category I structures. The Class 1E station batteries and associated dc distribution system are therefore protected from the applicable extreme external hazards. The FLEX DG enclosure also contains electrical distribution equipment, and it was designed to be robust for all applicable hazards. Licensee procedure 0EOP-01-SBO-10, "Battery Load Stripping," directs operators to conserve dc power during the event by stripping non-essential loads. The load shedding actions will only be performed if the permanently pre-staged FLEX DGs fail to start as planned. Because the FLEX DGs are permanently pre-staged, they are expected to be available prior to the time required for additional load shedding actions.

Each unit at BSEP has four Class 1E station batteries separated in two divisions, Division I (1A-1 and 1A-2, 2A-1 and 2A-2) and Division II (1B-1 and 1B-2, 2B-1 and 2B-2), in a 125/250 Vdc split dc system. The station batteries were manufactured by Exide Technologies. The Class 1E station batteries are model NCN-17 and consist of 60 cells that have a nominal 8-hour rating of 1200 ampere-hours.

The NRC staff reviewed the licensee's dc coping study, document 13-4085.001, "Unit 1 and 2 125/250 Vdc Battery Capability Study to Extended Loss of AC Power (ELAP)," Revision 0, which verified the capability of the dc system to supply the required loads during the first phase of the BSEP 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). The licensee noted and the staff confirmed that if it is necessary, the Class 1E station battery capacity could be extended up to 2.16 hours for battery 1B-2, and 2.25 hours for the remaining batteries by shedding non-essential loads.

Based on the 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 BSEP, Units 1 and 2, 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. This conclusion is based on the licensee's provisions that the pre-staged 480 Vac FLEX DGs are started and energize the battery chargers prior to the necessary batteries depleting to minimum acceptable voltage, and dc load shedding is completed (if required) within the times assumed in the licensee's analysis.

The licensee's Phase 2 strategy includes repowering selected 480 Vac buses within 1 hour after initiation of an ELAP event using one of the pre-staged 500-kW 480 Vac FLEX DGs. The pre-staged FLEX DG will supply power to emergency bus E6 and/or E8, energizing both units' Division II battery chargers. If both of the pre-staged FLEX DGs are available, different electrical lineups and loading may be utilized. The NRC staff reviewed licensee calculation 3116-CALC-E-001, "FLEX Diesel Generator Sizing Calculation," Revision 0, procedures 0EOP-01-FSG-04, "FLEX Diesel Generator Alignment," Revision 1, and 0EOP-01-SBO-07, "480V E-BUS Crosstie," Revision 1, conceptual single line diagrams, and the separation and isolation of the 480 Vac permanently pre-staged FLEX DGs from the Class 1E EDGs. In its FIP, the licensee stated that approximately 150 kW is required for the Division II loads from each unit, for a total required capacity of approximately 300 kW. Based on the margin available for the 500 kW 480 Vac permanently pre-staged FLEX DGs, the NRC staff finds that one FLEX DG has sufficient capacity and capability to supply the required electrical loads at both units for the licensee's Phase 2 strategies.

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

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 RPV 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 acceptable approach consisting of three 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; 2) makeup via connection to SFP cooling piping or other alternate location; and 3) spray via portable monitor nozzles from the refueling floor using a portable pump capable of providing a minimum of 200 gpm per unit (250 gpm if overspray occurs). During the event, the licensee selects the 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.

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. 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 other initial conditions; and Section 3.2.1.6 describes SFP initial conditions.

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.

3.3.1 Phase 1

The licensee stated in its FIP that no actions are required during Phase 1 for SFP makeup because the time to boil is sufficient to enable deployment of Phase 2 equipment. Adequate SFP inventory exists to provide radiation shielding for personnel well beyond the time of boiling. The licensee will monitor SFP water level using reliable SFPLI installed per Order EA-12-051.

3.3.2 Phase 2

In the licensee's FIP, Section 2.4.2, states that during Phase 2, operators will deploy one portable FLEX pump per unit to supply water from the CST or discharge canal to that unit's SFP. The FLEX pump discharge can be: (1) routed to a connection on the RHR fuel pool cooling assist line to the SFP cooling system (not requiring refueling floor access), (2) routed to the refuel floor to provide direct makeup to the pool, or (3) routed to spray nozzles to provide spray flow.

3.3.3 Phase 3

The FIP states that SFP cooling can be maintained indefinitely using the makeup strategies described in Phase 2 above. However, NSRC equipment is available during Phase 3 for SFP cooling and can provide additional defense-in-depth.

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) SFP cooling system is intact, including attached piping.

The licensee's FIP indicates that boiling begins at approximately 5 hours during a full core offload, outage situation. As described in the licensee's 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 staff noted that the FIP indicates that operators will open the Reactor Building roof hatch and doors within approximately 2 hours from event initiation to ensure the SFP area and the Reactor Building remains habitable for personnel entry.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves the use of a FLEX pump for each unit supplying water from the respective units' CST. The licensee's FIP indicates that the portions of the RHR and fuel pool cooling systems required for the strategy were designed for safety related service and thus are expected to be available following all postulated external events. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX pump is discussed in Section 3.7.3.1 below. Furthermore, the staff's evaluation of the robustness and availability of water sources for an ELAP event is discussed in Section 3.10.3. Based on these evaluations, the licensee's FIP statements, and the onsite audit walk down, the staff concludes that the SSCs supporting the SFP makeup strategy should be available.

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. The NRC staff's review of the SFPLI, 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

As described in Section 2.4.1 of the FIP, the SFP will boil in approximately 5 hours and boil off to a level requiring makeup 57 hours from initiation of the event with no operator action at the maximum design heat load.

In the BSEP UFSAR, Section 9.1.2.3.2.3, states that the two bounding scenarios analyzed are: (1) normal operation heat load and (2) the maximum normal/emergency refueling heat load which includes a full core offload. The heat loads, boil-off times, and makeup rates can be found in the table below.

	Heat Load	Time to boil	Makeup rate
Case 1	14 million Btu/hr	16 hrs	28.7 gpm
Case 2	35 million Btu/hr	5 hrs	65 gpm

Therefore, the licensee conservatively determined that a SFP makeup flow rate of at least 65 gpm will maintain adequate SFP level above the fuel 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, the staff finds the licensee has considered the maximum design-basis SFP heat load in the development of their strategy for SFP makeup under the postulated conditions.

3.3.4.3 FLEX Pumps and Water Supplies

As described in the FIP, the SFP cooling strategy relies on the FLEX pump to provide SFP makeup during Phase 2. In the FIP, Sections 2.3.10 and 2.4.7 describe the hydraulic performance criteria for the FLEX pumps and the EDMG pump. The staff notes that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3 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 pumps and EDMG pump can provide SFP makeup at a rate that exceeds the required flow rate or alternatively provide a SFP spray flow rate of 250 gpm. The staff concludes that the licensee's plan should provide SFP makeup via multiple options in accordance with the provisions of NEI 12-06, Revision 0. Furthermore, the staff finds that the licensee's analysis was developed consistent with NEI 12-06, Section 11.2 and the FLEX equipment should be available during an ELAP event.

3.3.4.4 Electrical Analyses

The NRC staff performed an analysis of the licensee's electrical strategies as described in Section 3.2.3.6 of this SE, which includes the SFP cooling strategy. The evaluation of the electrical aspects of the licensee's SFP level instrumentation (SFPI) system is contained in Section 4.0 of this SE.

Based on its review, the NRC staff finds that the licensee's strategy to restore or maintain SFP cooling indefinitely during an ELAP as a result of a BDBEE is acceptable.

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 some 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. Brunswick, Units 1 and 2, are BWRs with a Mark I containment. For this containment type, NEI 12-06 provides guidance that a reliable, hardened vent (or hardened containment vent system (HCVS)) may be used to remove heat and control pressure buildup inside the primary containment. One such approach is for a licensee to perform an analysis demonstrating that containment pressure control is not challenged.

The licensee performed a containment evaluation, BNP-MECH-FLEX-0002, "Brunswick Nuclear Plant Containment Analysis of FLEX Strategies," which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of venting containment through the hardened wetwell vent prior to reaching the PCPL of 70 psig, approximately 17.7 hours after the occurrence of the ELAP-inducing event. The UFSAR Table 6-3 design limits for pressure and temperature of the drywell containment are 62 psig and 300°F, respectively and the UFSAR Table 6-3 design limits for pressure and temperature of the suppression pool are 62 psig and 220°F, respectively. The licensee's calculation demonstrated that employing this venting strategy exceeds the design limits for temperatures and pressures in the drywell containment and the suppression pool. However, the analyzed peak pressures are maintained below the PCPL and have considerable margin to the drywell failure pressure. The drywell temperature temporarily exceeds the design limit at about 17 hours before returning beneath the limit. The peak temperature is less than the limit determined by BSEP for various structural components in containment (limiting value for all components is 350°F). A more detailed discussion for exceeding the design limits, specifically for the suppression pool temperature, is discussed in Section 3.4.4.1.1.

Based on review of the licensee's UFSAR, FIP, and FLEX containment evaluation, the NRC staff notes that the required actions to maintain containment integrity and required instrumentation functions have been developed, and are further summarized below.

3.4.1 Phase 1

During Phase 1, BSEP will depressurize the RPV to the range of 150 psig to 300 psig using SRVs, which direct steam into the suppression pool. This action will absorb the heat load in the primary containment, limiting the pressure rise and prolonging coping time. The SRVs are powered by the station batteries and the pneumatic motive force is nitrogen backup. The licensee's MAAP analysis (BNP-MECH-FLEX-002) predicts that, while following the core cooling strategy specified in the governing procedures, the suppression pool will heat up to 190°F approximately 3 hours after the initiation of the ELAP event, at which point the RCIC suction is re-aligned to the condensate storage tank.

With regard to necessary instrumentation to employ the strategies, the FIP states that the Phase 1 coping strategy for containment involves monitoring containment temperature and pressure using installed instrumentation. Specifically, it states that monitoring of containment drywell pressure and temperature will be available via normal plant instrumentation.

3.4.2 Phase 2

When the nitrogen backup cylinders are depleted, a FLEX air compressor will provide pneumatic motive force for the SRVs. As the suppression pool heats up due to blowdown from the SRVs and RCIC operation, the containment will begin to heat up and pressurize. The licensee's MAAP calculation tracks containment pressure to determine the approximate time that the HCVS will be opened to prevent exceeding the PCPL (70 psig). This occurs at approximately 17.7 hours after the event initiation. At this time, the HCVS will be opened in accordance with plant emergency operating procedures (EOPs) to provide containment heat removal and begin a long-term strategy of reactor water make-up and boiling to protect the core and containment. The opening of the HCVS will prevent any further rise in suppression pool water temperature. According to the licensee, the PCPL has considerable margin to the drywell failure pressure limit of 170 psig.

Containment pressure will be maintained below the PCPL as directed by procedure 0EOP-02-PCCP, "Primary Containment Control Procedure." Thus, containment pressure control is appropriately addressed by invoking this strategy. The HCVS can be powered by the permanently pre-staged FLEX DGs, which will be operating prior to containment reaching the pressure limit. Brunswick will continue venting through the HCVS, as necessary.

3.4.3 Phase 3

The licensee's FIP states that the Phase 3 strategy is a continuation of the Phase 2 strategy and that there is no defined end time for the Phase 2 coping period for maintaining containment integrity. The NRC staff notes that the NSRC will supply additional equipment (e.g., CTGs) and that in the long-term, the licensee's plan provides BSEP decision makers with the necessary indications to respond to any containment challenges.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

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 Structure

Section 6.2.1.1 of the BSEP UFSAR [Reference 45], describes the Mark I containment as a steel-lined, reinforced concrete structure which encloses the reactor vessel, the reactor coolant recirculation system, and other branch connections of the reactor primary coolant system. This section further states that the primary containment includes a drywell and a pressure suppression chamber connected by vents, isolation valves, vacuum breakers, cooling systems, and other service equipment. The primary containment is a Class I structure, and it is designed to withstand the jet forces resulting from a rupture of any reactor coolant system pipe.

Section 6.2.3 of the BSEP UFSAR states that the secondary containment system includes the containment structure (i.e. Reactor Building) and the safety-related systems provided to control the ventilation and cleanup of potentially contaminated volumes, which are exclusive of the primary containment, following a design basis accident. It states that the Reactor Building is designed in accordance with all applicable local and state building code requirements, and the seismic, wind, tornado, and missile design and loading criteria.

Furthermore, Section 3.2.1 of the BSEP UFSAR, lists the primary containment and secondary containment as being Class I structures which are designed to remain functional during and following the most severe natural phenomena which can be postulated to occur at the site.

According to the licensee's analysis, the UFSAR design limit for the suppression pool temperature (300°F) is calculated to be exceeded with a peak value of 305°F. According to the licensee's FIP, BSEP evaluated the structural components in containment and determined that the limiting value for all components would be 350°F. During the on-site audit, the staff reviewed BNP-PSA-046, "PSA Model Appendix J Containment Capability Analysis," Revision 0, to show the structural integrity of the suppression pool. This document assesses the BSEP containment structures associated with the Mark I containment design and the performance of the capability analyses for these structures under pressure and temperature conditions. The report concludes that the BSEP containments have been determined to have ultimate pressure capability reaching to the 200 to 220 psig range for temperatures of 350°F to 400°F. This represents a factor of 3.2 to 3.5 times the design accident pressure of 62 psig. According to the licensee, a pressure of 170 psig represents a level for which there is a high confidence of low probability of failure. This measure of capability reflects the primary shell structures, as well as all penetrations. Multiple potential failure modes might occur in this ultimate pressure range because of a relatively "balanced" design of many containment components. Principal among these modes are failures in the torus and in the drywell head. The analyzed peak pressures are below the PCPL and have considerable margin to the drywell failure pressure. The peak temperature is less than the limit determined by BSEP for various structural components in containment of 350°F. Therefore, the drywell and suppression chamber are not expected to fail.

Safety Relief Valves, Hardened Wetwell Vent, and Backup Nitrogen Supply

The FIP states that BSEP will utilize SRVs to depressurize the RPV and the hardened wetwell vent to maintain containment integrity. During an ELAP, these components are operated using dc power, ac power (from the FLEX DG's) and pressurized backup nitrogen. Brunswick calculations indicate that, accounting for potential actuations of the SRVs and other demands for compressed nitrogen (e.g., suppression pool vacuum breaker valves and hardened wetwell vent valves), the backup nitrogen supply has a capacity of at least 24 hours. The SRVs and the nitrogen supply are safety-related and located within the Reactor Building, which is a seismic category I structure and provides protection from all applicable hazards.

The HCVS vent path at BSEP consists of a wetwell vent on each unit. The wetwell vent exits the primary containment through the wetwell purge exhaust piping and associated inboard wetwell purge exhaust valve. Between the inboard and outboard wetwell purge exhaust valves, the wetwell vent isolation valve is installed. Downstream of the wetwell vent isolation valve, the vent piping exits the Reactor Building through the west wall and into the space between the Reactor Building and Turbine Building. The vent traverses up the exterior of the building and re-enters the Reactor Building through the metal siding on the refuel floor, then rises along the west side where it exits the Reactor Building through the roof. All effluents are exhausted above each unit's Reactor Building.

The FIP states that BSEP specifically evaluated the hardened wetwell vent pipe to confirm that it is robust to all hazards, including tornado missiles. It further states that the hardened wetwell vent is protected from all hazards. The FIP also states that the portions of the systems required for the FLEX strategy, including the HCV system and the related air/nitrogen systems were designed for safety related service and will be available following the applicable extreme external hazards. During the audit process the NRC staff reviewed calculation PGB-024-CALC-0002, "Evaluation of the Hardened Wetwell Vent for Beyond Design Basis External Events," Revision 0, to confirm the FIP statements.

Based on these UFSAR qualifications and FIP statements, the NRC staff finds that the containment, the HCVS, and the necessary support equipment credited in the strategy are robust, as defined by NEI 12-06, and should be available following an ELAP event induced by the postulated external hazards.

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 licensee's FIP states that BSEP will monitor the following parameters to support the FLEX containment integrity strategy. Associated instruments are initially powered by the station batteries and subsequently from the permanently pre-staged FLEX DGs.

- RPV pressure (narrow range)
- RPV pressure (wide range)
- Drywell temperature
- Drywell pressure
- Suppression pool water temperature
- Suppression pool water level

In the event of an ELAP, RPV pressure, drywell pressure and drywell/suppression pool temperature control room instrumentation would be available due to the coping capability of the station batteries and associated inverters in Phase 1, or the portable DGs deployed in Phase 2. If no ac or dc power was available, the FIP states that key credited plant parameters, including these containment parameters, would be available by the alternate means below:

- RPV pressure can be monitored from instruments directly on the instrument racks in the Reactor Building.

- Drywell pressure can be monitored by connecting a pressure test gauge to a drain valve in the Reactor Building.

- Drywell and suppression pool temperature can be monitored by connecting a Fluke meter to the Remote Shutdown Panel.

- Suppression pool water level can be monitored by connecting a pressure test gauge to a lower drain valve and connecting a pressure test gauge to a vent valve connected to the suppression pool air space. Utilizing these two readings the water level can be determined.

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 (BNP-MECH-FLEX-0002), which was based on the boundary conditions described in Section 2 of NEI 12-06. This calculation utilized the MAAP computer code to perform numeric computations of the fundamental thermodynamic equations which predict the heat up and pressurization of the containment atmosphere under ELAP conditions.

As stated in Section 3.4 above, the calculation concludes that the HCVS should be opened to begin removing heat and relieving pressure from the containment atmosphere prior to the containment pressure reaching 70 psig, approximately 17.7 hours after the occurrence of the ELAP-inducing event. It assumes that the reactor has been operating at 100 percent power for 100 days (as specified in NEI 12-06, Section 2) when the ELAP event occurs. During an ELAP event, the containment will begin to heat up and pressurize due to the discharge of the SRVs, leakage from the recirculation system (61 gpm), and the RCIC system exhaust steam as described in the core cooling strategy of Section 3.2.1. Under these conditions and with the employment of the heat removal strategy, the analysis indicated that the peak suppression pool water temperature is 305°F at a pressure of 69 psig and the drywell pressure peaks at 302°F at a pressure of 69 psig. As previously noted in this SE, each of these values exceed the UFSAR respective design limits for the containment parameters of pressure and temperature in the drywell and suppression pool. A justification for exceeding these design limits is discussed in Section 3.4.4.1.1. The analyzed peak pressures are below the PCPL and have considerable margin to the ultimate drywell failure pressure. The peak temperature is less than the limit determined by BSEP for various structural components in containment (limiting value for all components is 350°F). Therefore, the drywell and suppression chamber are not expected to fail.

3.4.4.3 FLEX Pumps and Water Supplies

As discussed in Section 2.5 of the BSEP FIP, permanently-installed plant equipment features are used to maintain containment integrity throughout the duration of the event; no non-permanently installed equipment (i.e., portable equipment) is required to maintain containment integrity.

3.4.4.4 Electrical Analyses

The licensee's Phase 1 coping strategy for containment involves monitoring key parameters (RPV pressure, drywell temperature and pressure, and suppression pool water temperature and level) using installed instrumentation and using SRVs powered by the Class 1E station batteries to depressurize the RPV.

The licensee's Phase 2 coping strategy uses the 480 Vac permanently pre-staged FLEX DGs to power the battery chargers, which will maintain dc bus voltage for continued availability of instrumentation needed to monitor key containment parameters. The 480 Vac permanently pre-staged FLEX DGs will also power the HCVS. The licensee's Phase 3 coping strategy is to continue the Phase 2 strategy with equipment supplied by the NSRC as a backup, if necessary.

Based on its review of licensee calculations as described in Section 3.2.3.6 of this SE, the NRC staff finds that the 480 Vac permanently pre-staged FLEX DGs should have sufficient capacity and capability to supply the required loads to reduce containment temperature and pressure, if necessary, to ensure that key components remain functional.

Based on its review, the NRC staff finds that the licensee's electrical strategy should support the necessary equipment to restore or maintain containment integrity and cooling indefinitely during an ELAP as a result of a BDBEE.

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-2012-0, 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; ice; 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 22] (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 46]. 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 43]. The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 23]. 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 damage 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 47]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 48]. 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 design-basis earthquake (DBE) or safe shutdown earthquake (SSE). As described in UFSAR Section 2.5.2, "Vibratory Ground Motion," the SSE seismic criteria for the site has a horizontal ground acceleration design value of 0.16g. 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 number 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 that the current design-basis for the limiting site flooding event is the probable maximum hurricane (PMH) event. As described in UFSAR Sections 2 and 3, the current design-basis is that the seismic category I structures are not susceptible to external flooding from the PMH or Probable Maximum Flood (PMF) Events. The most severe flood conditions are associated with a PMH coinciding with peak local astronomical tides.

The licensee also described in its FIP, that from the open coast, the surge water level propagation up the Cape Fear River into the intake canal was evaluated with a resultant peak level of 22 feet mean sea level (MSL). The nominal plant grade of 20 feet MSL results in two feet of water depth surrounding the plant during maximum surge conditions. All of the safety-related structures are waterproofed to elevation 22 feet MSL. For example, personnel and equipment access doors are provided with sills above the 22-foot still water level, or alternatively are equipped with positive seals and closure devices when the sills are below 22 feet MSL. The licensee's compliance letter dated May 19, 2016 [Reference 21], also indicates that the flood persistence above the site grade elevation of 20 feet MSL is expected to be between two and three hours.

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 miles per hour (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 per 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 U. S. NRC Regulatory Guide (RG) 1.76, "Design-Basis Tornado and Tornado Missiles For Nuclear Power Plants," Revision 1.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 33° 57' 30" north latitude and 78° 00' 30" west longitude. According to 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}", and Figure 7-2, "Recommended Tornado Design Wind Speeds for the 10^{-6} /yr Probability Level", the location of BSEP has a peak hurricane wind speed of 210 mph and a recommended tornado wind design speed of 200 mph. Based on the potential for winds in excess of 130 mph, the BSEP site is susceptible to damage from severe winds from a hurricane or tornado. Therefore, the plant screens in for an assessment for high winds associated with 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 33° 57' 30" north latitude and 78° 00' 30" west longitude. In addition, the site is located within the region characterized by EPRI as ice severity level 4 (NEI 12-06, Figure 8-2, "Maximum Ice Storm Severity Maps"). Consequently, the site is subject to severe icing conditions that could cause severe damage to electrical transmission lines. The licensee concludes that the plant screens in for an assessment for ice hazard.

In its FIP, the licensee also described that in accordance with NEI 12-06, snowfalls are unlikely to present a significant problem for deployment of FLEX in locations below the 35th parallel. Brunswick is located below the 35th parallel, so BSEP FLEX strategies are not required to consider the impedances caused by extreme snowfall. Guidance document NEI 12-06 also states that the same basic trend applies to extreme cold temperatures, so BSEP FLEX strategies are not required to address extreme cold temperatures.

Per BSEP UFSAR Section 2.3.2.1.2, "Temperature", the lowest temperature recorded was 0°F in December 1989 in Wilmington, North Carolina.

In summary, based on the available local data and Figures 8-1 and 8-2 of NEI 12-06, the plant site does experience ice storms; therefore, this hazard is screened in. The licensee has appropriately screened in the ice hazard.

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. The licensee stated that the extreme heat hazard is applicable for BSEP.

Per the BSEP UFSAR Section 2.3.2.1.2, the highest recorded temperature was 104°F in Wilmington, North Carolina in June 1952.

In summary, based on 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 that the majority of the FLEX equipment credited for Phase 2 strategies, and also cabling for Phase 3 electrical strategies, is stored in the FSB, which is a monolithic dome. The FSB is located on the southeast area of the plant next to the technical training center at an elevation above the maximum flood level for the site. As described in BSEP plant procedure 0PLP-01.4, "Fukushima Flex System Availability, Action, and Surveillance Requirements," Revision 1, the FSB has two available deployment doors capable of being opened by either manual or hydraulic means.

In its FIP, the licensee describes that the equipment stored in the FSB includes the following:

- Two FLEX pumps and one EDMG pump with couplings and hose trailers
- Spare nozzles and fittings for SFP spray makeup strategy
- Spare SFP level instrumentation (SFPI) batteries
- FLEX air compressors with equipment trailers to make pneumatic connections
- A CAT 924K loader and a Dodge 5500 flatbed truck for debris removal and equipment towing
- Salt spreader and salt bags for treating ice on deployment paths
- Diesel fuel oil (DFO) tank and trailer with transfer pump and hoses
- Discharge weir/canal suction trailer with hoses and fittings
- Control room ventilation trailer with exhaust fans, small portable DGs, and ducting
- NSRC generator cable trailer with cabling to connect the NSRC generators to the BSEP electrical distribution system

In addition, BSEP also has a FLEX DG enclosure. The FLEX DGs are permanently pre-staged in the FLEX DG enclosure. According to the licensee, the FLEX DG enclosure is protected from all external hazards applicable at the BSEP site.

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 described that the FSB was designed to withstand the seismic loading exceeding the American Society of Civil Engineers (ASCE) standard ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures," requirement for the Southport location and the BSEP SSE. According to the FIP, the FSB is designed to withstand two times the SSE without gross failure. In addition, the FSB was designed for combined loads as specified in ASCE 7-10 for dead loads, seismic loads, wind loads, wave loads, buoyancy loads, and snow and ice loads.

In Enclosure 2, "BSEP, Units 1 and 2, Interim Staff Evaluation Open and Confirmatory Items, and Audit Open Items", of the licensee's compliance letter [Reference 21], the licensee described that the FSB has seismically designed anchor points within the building. In addition, BSEP conducted an evaluation for the FLEX equipment storage configuration to be maintained in the FSB. The equipment is maintained in the FSB in accordance with this evaluation which uses spacing as the means for protecting large portable equipment during a seismic event, rather than securing large portable equipment. All stored equipment is placed in the building such that ample spacing between nearby adjacent portable equipment is maintained as a defense-in-depth measure such that if some equipment were to move during a seismic event it will not impact any other equipment. The actual locations of the portable equipment is marked on the dome floor so that its location can be controlled.

In its FIP, the licensee described that the FLEX DG enclosure was designed to withstand two times the BSEP SSE without gross failure. Additionally, the FLEX DG baseplate frame, the anchorage securing the permanently pre-staged FLEX DGs to the DFO tank vault (DFOTV), and the DG exhaust and vent piping are designed for two times the BSEP SSE.

In Enclosure 2 of the licensee's compliance letter [Reference 21], the licensee described that BSEP does not have any sources of large internal flooding that are not seismically robust nor any that will cause a flood upon loss of ac power. The large external sources of water for BSEP are from the ultimate heat sink, which is below the elevation of the site and requires a motive force to transport the water into the buildings. The internal areas for FLEX response include the EDG fuel oil vault, the EDG Building, the Unit 1 and Unit 2 Reactor Buildings, and the Control Building 23 foot elevation and above. None of these structures are subject to large internal flooding from external sources when ac power is not present. None of these areas rely upon ac power to mitigate ground water in critical locations.

3.6.1.2 Flooding

In its FIP, the licensee described that the FSB is located on the southeast area of the plant next to the technical training center at an elevation above the maximum flood level for the site.

The licensee also described in the FIP that the FLEX DG enclosure was constructed at an elevation to maintain FLEX DG equipment and electrical distribution system above maximum flood levels. The FLEX DG enclosure is also sealed against external flooding using a combination of welded seams/connections and sealant. The FLEX DG enclosure includes ventilation louvers that are designed to limit wind-driven rain intrusion.

In its FIP, as an additional protection item, the licensee described that they have designed cliff edge barriers that can be temporarily installed at certain entrances to structures housing safety-related equipment if a hurricane was approaching the BSEP site.

3.6.1.3 High Winds

In its FIP, the licensee described that the FSB is designed for a high wind loading exceeding the ASCE 7-10 requirement for the Southport location and the requirements found in ASCE paper No. 3269, "Wind Forces on Structures." In addition, the licensee stated in its FIP that the tornado loading is based on RG 1.76, revision 1 for wind speeds and tornado missiles.

The licensee also described in its FIP that the FLEX DG enclosure was designed to meet the requirements of ASCE 7-10 for hurricane and tornado wind loading and the range of tornado missiles specified in the BSEP UFSAR.

In Enclosure 2 of the licensee's compliance letter [Reference 21], the licensee described that an evaluation was completed that documents the CSTs are robust to all applicable external hazards with the exception of tornado missiles impacting the connection points on the CSTs (nozzles and associated piping). Missile barrier protection was therefore designed and installed for the CSTs such that the CSTs will be available for use following a BDBEE, including tornados.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee described that the FSB is designed for snow and ice loads on the roof based on the plant design-basis and ASCE 7-10. In addition, the FSB was designed for combined loads as specified in ASCE 7-10 for dead loads, seismic loads, wind loads, wave loads, buoyancy loads, and snow and ice loads.

The licensee also described in its FIP, that the FLEX DG enclosure was designed to meet the requirements of ASCE 7-10 for a roof live load of 30 pounds per square foot, which envelopes the applicable roof loads for BSEP associated with an ice storm.

Regarding FSB environmental conditions, the NRC staff reviewed Engineering Change Package 90400, "BNP Permanent FLEX Storage Building," Revision 0, during the audit process. According to the licensee's design package, the FSB's heating and ventilation system is designed to maintain a minimum temperature of 40°F and a maximum temperature of 110°F. In addition, the heating and ventilation system are designed for a range of external ambient temperatures from -9°F to 107°F. Humidity controls maintain the relative humidity inside the FSB at or below 70 percent.

As described in Enclosure 2 of its compliance letter [Reference 21], the licensee noted that equipment used in FLEX strategies will be procured as commercial equipment. Brunswick relies on portable fire pumps, air compressors, fuel oil transfer pumps, Dodge 5500 truck and a Caterpillar 924K, all of which were purchased as commercial grade equipment and will be staged as required outside of all buildings and/or structures during their use.

In Enclosure 2 of the licensee's compliance letter, the licensee described that no BSEP FLEX response components require heat tracing to implement the FLEX strategies. The only components of concern relate to the CST HPCI [High Pressure Coolant Injection]/RCIC level instrumentation. These instruments are not credited in the FLEX response.

3.6.2 Reliability of FLEX Equipment

Section 3.2.2 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.

Based on the number of portable FLEX pumps, permanently pre-staged 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 of NEI 12-06.

3.6.3 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.

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee described that the deployment path for FLEX equipment from the FSB is through site parking lots through the sally port gate, and then between the maintenance shop and service building to deployment areas. The deployment path is entirely over paved surfaces.

3.7.1 Means of Deployment

In its FIP, the licensee described that at the beginning of the event, BSEP will perform an initial assessment to determine accessibility of FLEX connections, conditions of deployment paths, and availability of equipment staging areas. Debris removal vehicles will be used as necessary to enable FLEX deployment.

In addition, the licensee described in its FIP that the CAT 924K loader is the primary equipment credited for clearing debris. Brunswick can also clear debris with a Dodge 5500 flatbed truck, if necessary. The Dodge 5500 flatbed truck is the primary equipment credited for towing FLEX equipment into deployment locations. The CAT 924K loader can also be used for towing FLEX equipment.

As described in Enclosure 2 of its compliance letter, for events caused by extreme ice, the CAT 924K may be used to remove ice in the travel path, but the Dodge 5500 and the salt spreader attachment will be the primary means of controlling ice buildup on the deployment pathways.

3.7.2 Deployment Strategies

In its FIP, the licensee described that analysis supporting Section 2.5.4.8, liquefaction potential, of the UFSAR has shown that soil liquefaction will not occur in the plant area under dynamic loadings of the SSE. Additional analyses performed assuming an earthquake of two times the SSE indicated liquefaction of loose sands could result in settlement of the ground surface outside of main plant areas but within the planned deployment route from the FSB. However, BSEP concluded that the potential liquefaction and settlement would not result in abrupt ruptures or significant offsets of paved surfaces, so there would not be a disruption to deployment of FLEX deployment.

In its FIP, the licensee described that the flooding hazard does not adversely impact the deployment strategy for moving FLEX equipment from the FSB to the planned deployment locations inside the protected area. The deployment paths are completely paved, as is most of the site. The paved areas facilitate efficient draining from the site proper to the intake and discharge canals and the surrounding creek systems such that movement of equipment from the FSB can be supported as early as two hours after site flooding above grade. There are no deployment requirements this early in the event, and based on the 2-3 hour persistence of the PMH-based flood above the site grade, the licensee concludes that deployment would not be hindered.

As described previously, the FLEX DGs are permanently pre-staged in the FLEX DG enclosure, therefore movement of the FLEX DGs is not required. As described in the FIP, the FLEX pump(s) can be deployed to the east side of the augmented off gas building (Unit 1) and/or to the south side of the Unit 2 CST. Suction hoses will be connected from CST FLEX connections to the intake of the FLEX pump(s). If the source of makeup water is the discharge canal, the FLEX pump can be deployed to the discharge weir. In this case, strainers will be placed on the suction hoses to reduce intake of debris.

Several paths for hose deployments have been identified to ensure a water supply for RPV and SFP makeup. Accordingly, BSEP will be able to deploy all components required for water distribution following any applicable BDBEE. As discussed previously, the paved areas needed for FLEX deployment drain efficiently. Locations for pump deployment and hose runs will not be impacted by flooding.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling

The licensee described in its FIP, that the diesel-driven FLEX pump suctions can be connected to the CSTs or placed in the discharge canal. As described in Section 2.3.4.7 of the FIP, both CSTs are protected from all applicable hazards. The FLEX pumps can be connected to the CSTs via a connection located 1.5 feet off the ground within the CST tornado missile barriers.

For core cooling, the FLEX pump discharge can be routed to a primary FLEX RPV makeup connection and an alternate FLEX RPV makeup connection. For the primary connection, the licensee can connect the FLEX pump discharge to the exterior portion of a penetration through the Reactor Building wall. The licensee then plans to connect another hose from the interior of the Reactor Building penetration to the reactor water clean-up system, which connects to the RCIC system, feedwater system, and to the RPV. The licensee stated in Section 2.3.4.2 of the FIP that the reactor water clean-up system, feedwater system, and RCIC system were designed for safety-related service and will be available following an BDBEE. As outlined in FIP Section 2.3.5.1, the exterior portion of the flow path has a protective enclosure providing protection from all applicable external hazards. Since the internal portion of the flow path is located in the Reactor Building, it too is protected from all applicable hazards. Additionally, the alternate connection is located on the exterior of the Reactor Building and connects to the ILRT system, part of the service air system, which can be connected to the RHR system and thus to the RPV. Although the service air system is not safety-related, the licensee states in Section 2.3.4.2 of the FIP that all portions of the service air system utilized for FLEX satisfy the requirements of NEI 12-06 for alternate injection paths. The licensee's FIP system listing indicates that the primary path is fully protected from all applicable external hazards.

The licensee stated in Section 2.3.5.3, that the backup nitrogen supply can operate the SRVs for at least 24 hours following a BDBEE. In addition, the licensee has primary and alternate pneumatic connections for diesel-driven FLEX air compressors to support the continued operation of the SRVs. The primary connection is located in a seismic isolation space between the Reactor Building and the Turbine Building. As stated in the FIP, the primary connection is located in the non-interruptible instrument air system and is robust for all hazards except for wind-generated missiles on three sides. The alternate connection is protected from all applicable hazards, except seismic, because it located in the Reactor Building, and it connects to the reactor non-interruptible air system. Thus, the NRC staff concludes that the air connections provide reasonable assurance of at least one connection being available for all postulated events.

Spent Fuel Pool Cooling

In the FIP, Section 2.4.4.5 describes the licensee's SFP makeup strategy connections. The licensee has two independent flow paths for providing SFP make up from the same FLEX pumps that are used to provide core cooling. The primary flow path utilizes the same Reactor Building penetration and exterior connection as the primary core cooling flow path described above. Once inside the Reactor Building, a gated wye is used to split off flow to the RHR system, which can be cross-connected with the fuel pool cooling and cleanup system. This primary connection does not require access to the refueling floor. The alternate flow path is to route hoses through the Reactor Building personnel access to a gated wye. From the gated wye, hoses can be routed up to the refueling floor and directly into the pool or through portable spray nozzles to provide spray flow.

Conclusion

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 core and SFP cooling via a portable pump during an ELAP caused by an external event.

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 strategy for supplying power to the necessary equipment will be accomplished by routing power from one of the two permanently pre-staged 480 Vac FLEX DGs located in the FLEX DG Enclosure, which was designed to be robust for all applicable extreme external hazards. The pre-staged 480 Vac FLEX DGs will supply power to emergency bus E6 and/or E8, energizing both Units' Division II battery chargers. Associated cabling is permanently pre-staged so deployment only consists of racking in a 480 Vac breaker at emergency bus E6 and/or E8 and starting the permanently pre-staged FLEX DGs. Only one of the two 480 Vac permanently pre-staged FLEX DGs is required to support the FLEX strategies for both units and each FLEX DG has an independent FLEX DG output distribution panel in the FLEX DG Enclosure. One FLEX DG output distribution panel connects to emergency bus E6 and the other connects to emergency bus E8. The two FLEX DG output distribution panels can be cross-tied so that one FLEX DG can supply power to both buses E6 and E8, thereby supporting operation of all key electrical equipment. If both 480 Vac permanently pre-staged FLEX DGs are placed in service, emergency bus E5 can be cross-tied to E6, and E7 can be cross-tied to E8, allowing the use of Division I battery chargers and Control Building HVAC. During the audit process, the NRC staff observed that procedure 0EOP-01-FSG-04 provides guidance for connecting the pre-staged 480 Vac FLEX DGs and procedure 0EOP-01-SBO-07, provides guidance for cross-tying E5 to E6 and E7 to E8. During the audit process, the licensee stated that phase rotation checks for the pre-installed FLEX DGs were conducted during installation testing.

For Phase 3, the licensee will receive four (two per unit) 1 MW 4160 Vac and two (one per unit) 1100 kW 480 Vac CTGs from an NSRC. The licensee plans to only connect the 480 Vac CTGs and not the 4160 Vac CTGs. In its FIP, the licensee stated that the CTGs supplied by an NSRC have an output connection that is not directly compatible with the electrical distribution system at BSEP. To protect the installed plant emergency electrical distribution system, BSEP maintains cables and connectors that are stored in the BSEP FSB to connect the 480 Vac CTG. The cabling approach for the 480 Vac CTG includes connections inside the FLEX DG Enclosure, which provides protection from all applicable external hazards. The 480 Vac CTGs will be staged on a concrete pad near the auxiliary boiler house. During the audit process the NRC staff reviewed procedure 0EOP-01-FSG-09, "FLEX NSRC Generator Operation," Revision 1, and confirmed that it provides guidance for connecting and verifying proper phase rotation before attempting to power equipment from the 480 Vac CTGs.

Based on its review of the licensee's FIP, electrical diagrams and station procedures, the NRC staff finds that the licensee's approach is acceptable given the protection and diversity of the power supply pathways, the separation and isolation of the permanently pre-staged FLEX DGs from the Class 1E EDGs, and availability of procedures to direct operators how to align, connect, and protect associated systems and components.

3.7.4 Accessibility and Lighting

In its FIP and Enclosure 2 of the compliance letter [Reference 21], the licensee described that the flashlights carried by operations personnel during normal operations are the credited FLEX strategy. In addition to this credited lighting, BSEP has the following additional equipment staged in the FSB:

- Spare flashlights and batteries
- Headlamp flashlights
- DC-powered portable light emitting diode (LED) units (36 portable lightweight LED lights and 22 larger LED lights)
- Diesel-powered light towers (two diesel light towers for outdoor lighting/4,000 watts)

Also described in the FIP is that throughout the station, including in the control room, there is lighting powered by dc distribution or from emergency light units likely to be available to supplement portable lighting. As part of Phase 3, the NSRC will deliver additional diesel-powered mobile lighting towers to augment BSEP lighting equipment.

3.7.5 Access to Protected and Vital Areas

In Enclosure 2 of its compliance letter [Reference 21], the licensee described that the access to the protected areas that may be affected by loss of ac power are security doors. Security doors have key-locks that allow access/egress upon a loss of the security function. Specified operations personnel have access to security keys that can override the loss of door security function.

During the audit process, the licensee provided information describing how access to protected areas will not be hindered. The NRC staff concludes that 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 2.8.3, the licensee described that each of the four EDGs at BSEP has a fuel oil storage tank. Each tank is located in an underground vault that is protected from all applicable hazards and is a seismic category I structure. Each tank has a minimum supply of 22,650 gallons of fuel controlled by the plant TSs. Brunswick will deploy a FLEX fuel trailer to the ground above one of the EDG tanks, open a penetration flange and connect a suction hose to the selected underground tank. Brunswick will then use a transfer pump to fill a 1,240-gallon trailer, which will be used to refuel portable equipment. The fuel trailer can be pulled by the CAT loader or the Dodge 5500 flatbed truck, both of which are stored in the FSB. The transfer pump can also be used to directly refuel the FLEX DGs, which are permanently pre-staged in the FLEX DG enclosure. Based on the design and location of these EDG fuel tanks and protection, the staff finds the tanks are robust and the fuel oil contents should be available to support the licensee's FLEX strategies during an ELAP event.

As stated above, the EDG fuel oil storage tanks have approximately 90,000 (4 x 22,650) gallons total between all four tanks. As part of the answer to OIP open item 3.2.4.9.A (found in Enclosure 2 to the licensee's compliance letter dated May 19, 2016), the licensee calculated that the 72-hour total consumption for all FLEX equipment is 10,167 gallons. Given the relatively small fuel demand for the Phase 2 FLEX components cited above and the large amount of additional fuel, BSEP has a sufficient inventory of fuel for diesel-powered equipment required for the FLEX strategy's until additional fuel arrives from off-site including the potential additional consumption of the Phase 3 equipment. Furthermore, the staff finds that the ability to refuel the diesel-powered FLEX equipment should ensure uninterrupted operation to support the licensee's FLEX strategies.

In the licensee's FIP, Section 2.8.3 states that all FLEX equipment will be stored with a fuel tank at least 3/4-full. The FIP did not explicitly state how the fuel stored in the FLEX equipment would be maintained. However, Enclosure 1 of the licensee's compliance letter [Reference 21], states that the licensee will follow the EPRI guideline for the FLEX equipment maintenance and test program. The NRC staff reviewed the endorsed EPRI guideline and confirmed that it recommends performing fluid analysis to check for age and contamination issues. Based on the above, the NRC staff finds that the FLEX equipment should be properly maintained (including fluids) and available at the start of a BDBEE.

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 BSEP 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, one located near Memphis, Tennessee and one near 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 24], 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 BSEP Alternate Staging Area D is not used. Staging Area C is the Wilmington International Airport. Staging Area B is an on-site parking lot (west parking lot). Staging Area A is the plant location of need. Use of helicopters to transport equipment from Staging Area C to Staging Area B is recognized as a potential need within the BSEP SAFER Plan and is provided for.

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 BSEP, ventilation that provides cooling to occupied areas and areas containing required equipment will be lost. The primary concern with regard to ventilation is the heat buildup that 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 related to FLEX implementation to ensure the environmental conditions remain within equipment design limits for the equipment to remain functional. The key areas identified for all phases of execution of the FLEX strategy activities are RCIC pump room, control room, battery room (including battery chargers and inverters), and containment.

RCIC Pump Room

The NRC staff reviewed licensee calculation BNP-MECH-FLEX-001, "FLEX Reactor Building Gothic Heat Up Analysis," Revision 0, which modeled the RCIC pump room during an ELAP event. In the FIP, the licensee stated that the expected room temperature will reach 110°F in 6 hours and a maximum temperature of 131°F in 7 days if the Reactor Building roof hatch and doors are opened within 6 hours into an ELAP event. The maximum temperature limit for equipment functionality in the RCIC pump room area is 165°F. During the audit process, the staff reviewed the licensee's procedure 0EOP-01-SBO-04, "Blacked Out Unit Local Actions," Revision 1, which provides guidance to open the Reactor Building roof hatch and doors.

Based on the expected temperatures remaining less than the acceptable room limits and on the licensee's mitigating actions, the NRC staff finds that the required electrical equipment in the RCIC pump rooms should not be adversely impacted by a loss of ventilation as a result of an ELAP event.

Control Room

The licensee's FIP indicates that the licensee has a strategy to open control room doors and provide supplemental control room ventilation using portable ducting, fans, and small diesel generators from the FSB. These actions will limit the control room temperature rise to a peak of 116°F with the longer-term temperature dropping to approximately 100°F. In addition, should both the "N" and "N+1" permanently pre-staged FLEX DGs be available, the licensee would power the Control Building HVAC system to provide cooling in lieu of using the supplemental cooling strategy. During the audit process, the NRC staff reviewed licensee report, RWA-L-1312-003, "Control Building FLEX Room Heat-up Analysis for the Brunswick Nuclear Plant," dated February 15, 2015, which modeled the control room during an ELAP event to confirm the analytical basis for the FIP statements. In addition, the NRC staff reviewed procedure 0EOP-01-SBO-02, "Blacked Out Unit Initial Actions," Revision 1, which provides guidance to open control room panel doors open within 30 minutes and restore normal control ventilation within 4 hours. This procedure also provides guidance to establish alternate control room ventilation if normal ventilation cannot be restored.

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 control room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Vital Battery Room

The licensee's FIP indicates that, should only one FLEX DG be available, doors to the battery rooms will be opened to establish a ventilation flow path. In addition, should both the "N" and "N+1" FLEX DGs be available, the licensee would power the normal battery room ventilation system. In order to validate this strategy with respect to maximum potential equipment temperatures, during the audit process the NRC staff reviewed the licensee's report, RWA-L-1312-003, which modeled the vital battery rooms during an ELAP event. The report assumed that the initial battery room temperatures were 96°F (Unit 1 battery room "A"), 97°F (Unit 1 battery room "B"), 95°F (Unit 2 battery room "A"), 97°F (Unit 2 battery room "B") and the expected maximum temperature in the Unit 1 and Unit 2 battery rooms was calculated to reach 120°F and 119°F, respectively if the necessary doors are opened within one hour. The battery vendor's (Exide Technologies) analysis shows that the station batteries are capable of performing their function up to 125°F, however at this temperature, periodic monitoring of electrolyte level may be necessary to protect the battery since the battery may gas more at higher temperatures. The report conservatively did not take into account restoring normal battery room ventilation when power is available via the permanently pre-staged Phase 2 FLEX DGs. The staff also reviewed procedures 0EOP-01-FSG-04 and 0EOP-01-SBO-07, which provide guidance to restore normal battery room ventilation when the permanently pre-staged FLEX DGs are powering the appropriate 480 Vac bus and the station batteries are charging. If the normal battery room ventilation cannot be placed in service, the licensee's procedures provide guidance to secure open the necessary doors, consistent with the analysis.

Based on the above, the NRC staff finds that the licensee's ventilation strategy, in combination with the effect of the heat sinks in the dc equipment rooms, will maintain the battery room temperature below the maximum temperature limit (125°F) of the batteries, as specified by the battery manufacturer. Therefore, the NRC staff finds that the BSEP vital batteries should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP event.

Containment

See Section 3.4.4.4 of this SE for the NRC staff's evaluation of the licensee's mitigating strategy for maintaining containment temperature within the design limits of credited instrumentation and electrical equipment.

3.9.1.2 Loss of Heating

In its FIP, the licensee stated that the only components of concern include the CST HPCI/RCIC level instrumentation. Heat tracing for this instrumentation is required to maintain operability of the level instruments and the heat trace is assumed to be in service preventing freezing of the associated piping/instrumentation at the time of the BDBEE. These instruments are not credited in the FLEX response.

Brunswick is below the 35th parallel, therefore per Section 8 of NEI 12-06, snow or extreme cold hazard conditions do not apply to the BSEP site. Therefore, the NRC staff concludes that protection of the battery rooms for extreme low temperatures should not be required for implementation of the BSEP FLEX strategy.

3.9.1.3 Hydrogen Gas Accumulation in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3, is the potential buildup of hydrogen in the vital battery rooms as a result of loss of ventilation during an ELAP event. During the audit process, the NRC staff reviewed licensee calculation, 0FP-0001, "Battery Room Hydrogen Generation," Revision 2, to verify that hydrogen gas accumulation in the 125 Vdc vital battery rooms will not reach combustible levels while HVAC is lost during an ELAP. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. In its FIP, the licensee stated that the hydrogen mixture in the battery rooms would not reach the explosive limit for at least 8.4 hours without ventilation. Procedures 0EOP-01-FSG-04 and 0EOP-01-SBO-07, provide guidance to restore battery room ventilation when the permanently pre-staged FLEX DGs are powering the 480 Vac bus and the Class 1E station batteries are charging. If battery room ventilation cannot be placed in service, the licensee's procedures provide guidance to open the necessary doors.

Based on its review of the licensee's calculation and battery room ventilation strategy, the NRC staff finds that hydrogen accumulation in the BSEP 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 licensee performed a GOTHIC analysis for the control room which showed that the maximum temperature reached would be 116°F. It is expected that building ventilation flow can be re-established and will subsequently reduce the control room temperature to approximately 100°F. Based on the licensee being able to establish control room temporary ventilation and maintain long-term control room temperatures below 110°F (the temperature limit, as identified in NUMARC-87-00, for personnel habitability), the NRC staff finds that personnel in the control room will not be adversely impacted by the initial loss of ventilation as a result of an ELAP event.

3.9.2.2 Spent Fuel Pool Area

See Section 3.3.4.1.1 above for the detailed discussion of ventilation and habitability considerations in the SFP area. In general, the licensee has established actions early in the event so that habitability is not required during anticipated boiling of the SFP. The licensee also has the ability to add water to the SFP from the installed SFP cooling piping without accessing the refueling floor.

3.9.2.3 Other Plant Areas - RCIC Room

In the FIP, Section 2.3.1 indicates that the Phase 1 core cooling FLEX strategies rely on the RCIC pump as the motive force for providing water to the RPV. The NRC staff noted that operator access to the RCIC pump may be necessary to reset the pump and/or take manual control. The staff's evaluation of the room temperatures for the RCIC compartment is documented in Section 3.9.1.1 of this SE.

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 3.2.2.5 states that cooling and makeup water inventories are considered available if they are contained in systems or structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles. The NRC staff reviewed the licensee's planned water sources to verify that each water source was robust as defined in NEI 12-06.

3.10.1 RPV Make-Up

Phase 1

As described in the FIP, the CST is the normal suction for the RCIC pump. The FIP also states that operators will align RCIC to the suppression pool for a portion of the sequence of events. As stated in FIP Section 2.3.4.7, both CSTs are protected from all applicable hazards. Furthermore, FIP Section 2.3.4.6 states that the suppression pool is located in the Reactor Building and is protected from all applicable hazards. Therefore both credited water sources for RCIC are available for all postulated external hazards.

Phase 2

During Phase 2, the licensee will transition from the RCIC pump to a portable FLEX pump to provide makeup water to the RPV. The robust water sources for the FLEX pump include the CST (described above) and the discharge canal. The FIP Section 2.3.4.8 states that the discharge canal will not be adversely impacted by any applicable external hazards. The NRC staff notes that other, non-robust, sources of water may be available depending on the event and, if so, would be used prior to the discharge canal as they provide cleaner water. These other sources include the demineralized water tank, the fire water tank, and condensate hotwells.

Phase 3

For Phase 3, RPV makeup strategy is the same as the Phase 2 strategy, with the consideration that if discharge canal water is being used, that the licensee would restore on-site capabilities for providing clean water make-up or arrange for an offsite water source.

3.10.2 Suppression Pool Make-Up

In its FIP, the licensee did not describe any provisions for suppression pool makeup. The NRC staff reviewed the licensee's MAAP analysis which shows suppression pool level slowly increasing during the event. The staff concludes that the licensee's strategy does not require suppression pool makeup and that based on the slow increase, there is sufficient time for the licensee to address an increasing level, should it be required.

3.10.3 Spent Fuel Pool Make-Up

Phase 1

No makeup is required in Phase 1.

Phase 2

Phase 2 makeup to the spent fuel pool is from the CST or the discharge canal via one FLEX pump per unit. The water sources for the FLEX pumps are described above in Section 3.10.1.

Phase 3

In its FIP, the licensee noted that the Phase 3 SFP cooling strategy is the same as the Phase 2 strategy.

3.10.5 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 the plant at power, the mitigation strategy initially focuses on the use of the steam-driven RCIC pump to provide the water initially needed for decay heat removal. If the plant has been shut down and all or most of the fuel has been removed from the RPV and placed in the SFP, there may be 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. The licensee's analysis shows that following a full core offload to the SFP, approximately 57 hours are available to implement makeup, and the licensee's FIP states that this provides ample time to deploy the makeup strategy.

When a plant is in a shutdown mode in which steam is not available to operate a steam-powered pump such as RCIC, another strategy must be used for decay heat removal. By letter dated September 18, 2013, NEI submitted to the NRC a position paper entitled "Shutdown/Refueling Modes" [Reference 38], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 39], the NRC staff endorsed this position paper as a means of meeting the requirements of the order.

The position paper 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. The NRC's endorsement letter concluded that the position paper provides an acceptable approach for demonstrating that the licensees are capable of implementing mitigating strategies in shutdown and refueling modes of operation. By letter dated May 19, 2016 [Reference 21], the licensee informed the NRC staff of its plans to follow the guidance in this position paper. 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 describes that the inability to predict actual plant conditions that require the use of BDBEE equipment makes it impossible to provide specific procedural guidance. As such, BSEP has added content to existing EOPs and developed new FSGs to provide guidance that can be employed for a variety of conditions. The EOPs and FSGs, to the extent possible, provide pre-planned FLEX strategies for accomplishing specific tasks in support of emergency response. The new FSGs are used to supplement (not replace) the existing procedure structure that establishes command and control for the event.

In addition, the FIP describes that the EOPs are the command and control structure when entering a plant emergency. FLEX strategies are embedded directly in the EOPs or are referenced by the EOPs.

3.12.2 Training

In its FIP, the licensee stated programs have been established to develop personnel proficiency in the mitigation of BDBEEs. The Systematic Approach to Training (SAT) process was utilized to analyze, design, develop and implement training for applicable personnel.

The FIP described that initial training was provided and continuing periodic training will be provided to site emergency response leaders on BDBEEs 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. The FIP also noted that FLEX drills will be scheduled and conducted.

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 compliance letter (Enclosures 1 and 2), the licensee stated that they would conduct maintenance and testing of the FLEX equipment in accordance with the industry letter.

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 0

3.14.1 Permanent Pre-staging of FLEX DGs

During the audit process, the NRC staff determined that the FLEX strategy to use permanently pre-staged DGs should be considered an alternative to the guidance in NEI 12-06, Revision 0. The licensee evaluated its strategy regarding permanently pre-staging two 480 Vac, 500 kW FLEX DGs in a single building (FLEX DG Enclosure) that is protected with respect to all extreme external hazards. A description of the licensee's power configuration regarding the permanently staged FLEX DGs is contained in Section 3.7.3.2 of this SE. The licensee provided a justification for the proposed alternative in a position paper submitted to the NRC by letter dated June 3, 2015 [Reference 20].

The licensee's evaluation considered the potential causes of a failure of both FLEX DGs and determined that such a failure is highly unlikely given the design and layout of the FLEX DG installation and the quality of the installed components. The evaluation included consideration of the potential for: (a) catastrophic mechanical failure of one FLEX DG potentially impacting the other, (b) inadvertent fire suppression system discharge in the FLEX DG Enclosure, and (c) fire in the FLEX DG Enclosure affecting both FLEX DGs.

The NRC staff reviewed the licensee's evaluation of the pre-installed FLEX DG configuration, specifically the reliability, design features, and hazard evaluation described in the FIP and the proposed alternative. Given due consideration to the overall strategy benefits of using pre-installed DGs, along with the licensee's justification for the proposed alternative, the NRC staff concludes that the permanent installation of the two FLEX DGs within a single enclosure should provide sufficient DG reliability to support the licensee's FLEX strategy. Therefore, the NRC staff finds the licensee's alternative to the NRC endorsed guidance provided in NEI 12-06 to be acceptable.

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 25], the licensee submitted its OIP for BSEP in response to Order EA-12-051. By letter dated May 23, 2013 [Reference 26], the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response by letter dated July 22, 2013 [Reference 27]. By letter dated November 18, 2013 [Reference 28], the NRC staff issued an ISE and RAI to the licensee. By letter dated March 31, 2015 [Reference 19], the NRC issued an audit report on the licensee's progress.

By letters dated August 26, 2013 [Reference 29], February 27, 2014 [Reference 30], August 28, 2014 [Reference 31], February 27, 2015 [Reference 32], August 26, 2015 [Reference 33], and February 24, 2016 [Reference 34], the licensee submitted 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 SFPLI, which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letters dated June 3, 2015 (Unit 2), and May 19, 2016 (Unit 1) [References 20 and 21], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved for BSEP.

The licensee has installed a SFPLI system designed by AREVA Americas, Inc. During a vendor audit the NRC staff reviewed AREVA's SFPLI system design specifications, calculations and analyses, test plans, and test reports. The staff issued a report describing the results of the vendor audit on September 15, 2014 [Reference 35].

The NRC staff also performed an onsite audit at BSEP to review the implementation of SFPLI related to Order EA-12-051. The scope of this audit included verification of whether the: (a) site's seismic and environmental conditions are enveloped by the equipment qualifications, (b) equipment installation met the vendor's requirements and recommendations, and (c) program features met the order's requirements. By letter dated March 31, 2015 [Reference 19], the NRC issued an audit report on the licensee's progress toward implementation of the order.

4.1 <u>Levels of Required Monitoring</u>

In its letter dated July 22, 2013 [Reference 27], the licensee identified the SFP levels of monitoring as follows:

- Level 1 is 37 feet 6 inches (116 feet 1 inches plant elevation). The minimum water level required in the skimmer surge tank, with an inlet temperature of 200°F, for the fuel pool cooling pumps to maintain their minimum NPSH is at plant elevation 94 feet. Section 3.7.7 of BSEP's TSs states that the SFP water level shall be > 19 feet 11 inches (115 feet 8 ¾ inches plant elevation) over the top of the irradiated assemblies seated in the spent fuel storage racks. The normal operating SFP water level is 37 feet 9 inches from the bottom of the SFP (116 feet 4 inches plant elevation). During normal operation, the weir for the skimmer surge tanks and SFP is positioned to maintain the level at or above the minimum value of 37 feet 6 inches. This level provides sufficient makeup to the skimmer surge tank to maintain skimmer surge tank water level at 94 feet plant elevation, thereby ensuring adequate fuel pool cooling pump NPSH.

- Level 2 is 26 feet 8 inches (105 feet 3 inches plant elevation). The highest point of any fuel rack in the SFP is approximately 16 feet 8 inches (95 feet 3 inches plant elevation).

- Level 3 is 16 feet 8 inches (95 feet 3 inches plant elevation).

The NRC staff review notes that the Level 1 designation is adequate for normal SFP cooling system operation and it is also adequate to ensure the required fuel pool cooling pump NPSH. This level also represents the higher of the two points described in NEI 12-02 for Level 1. The

Level 2 designation uses the first of the two options described in NEI 12-02 for Level 2, which is approximately 10 feet above the top of the fuel rack. The Level 3 designation is the top of the fuel rack.

The NRC staff finds that the licensee's proposed Levels 1, 2, and 3 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 Evaluation of Design Features

Order EA-12-051 required that the SFPLI shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Below is the staff's assessment of the design features of the SFPLI at BSEP.

4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the instrumentation will consist of two separate permanent fixed instrument channels per pool to monitor the SFP water level continuously, from normal water level (approximately 116 feet 4 inches plant elevation) down to a level at the highest point of any fuel racks [Level 3].

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 the measurement range for both of its SFPs, 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 OIP, the licensee stated that the level instrument channels will be installed in diverse locations and physically 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 SFP. The sensing component of each level instrument channel will be installed separately within the SFP in order to reduce common susceptibility to missiles and other external events. Electronics associated with the level instrumentation will be located outside the SFP operating area due to sensitivity to radiation. Cable routings will be installed that will provide reasonable protection from missiles that may result from damage to the structure over the SFP and refuel floor.

In its letter dated May 19, 2016 [Reference 21], the licensee provided sketches depicting the locations of the Unit 2 primary [Channel A] and back-up [Channel B] SFP level sensors, and the routing of the cables that will extend from the sensors toward the display location. In the same letter, the licensee stated that the Unit 1 waveguide pipe, sensors, and displays locations are similar to Unit 2. For each unit, both channels are mounted on the north edge of the SFP and are separated by a distance of 36 feet and 4-7/8 inches. The primary and back-up channels both route to adjacent east and west stairwells respectively and proceed vertically to the elevation below.

For the cabling routing of the SFP level instrument, the NRC staff reviewed the applicable licensee drawings, including those in the list below, and verified them during a walk down conducted during the onsite audit:

- F-02505, SK-89578-E-3001, "Unit 2 WR SFPLI (Ch. A or NE SFP Area) Instrument Location and Conduit Routing Plans," Revision B
- SK-89578-E-3002, "Unit 2 WR SFPLI (Ch. B or NW SFP Area) Instrument Location and Conduit Routing Plans," Revision A
- SK-89578-E-3003, "Unit 2 WR SFPLI (Ch. A & B) Control Room Instrument Location and Conduit Routing Plans," Revision B

Based on the drawing review, with verification by walk down during the onsite audit, the NRC staff concludes that there is sufficient channel separation within the SFP area between the primary and back-up level instrument channels 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 arrangement for the SFP level instrument 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 OIP, the licensee stated that each permanently installed instrument channel will be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the SFP structure.

By letter dated May 19, 2016 [Reference 21], the licensee further stated that the primary and back-up channels, which consist of the electronic sensor/transmitter, horn and waveguide piping, are mounted seismically. The mount designs for the electronic sensor support, horn support and intermediate supports were qualified considering total weight of the waveguide piping and its components and the seismic accelerations for the building structure. To meet the design criteria for a beyond-design-basis (BDB) event, the loading for the mounting supports were generated using a minimum of two times SSE seismic accelerations. Comparisons have been made to ensure the two times SSE accelerations were adequate for plant locations where the SFPLI equipment is installed. The fundamental frequency of installed systems was also calculated to ensure they were 20 Hertz (Hz) or greater. The mounting designs for these supports are qualified per calculations using the Manual of Steel Construction, AISC 8th Edition and 9th Edition, Hilti Product Technical Guides, and site specific specifications.

All anchorages are qualified using four concrete anchor bolts and the manufacturer's design guide. The generic calculation qualifies a simple C-channel steel section welded centrally on a 1/2 inch steel base plate. The base plate is anchored using four concrete anchor bolts. The mount calculation assumes generic seismic accelerations of 10g (horizontal) and 6.67g (vertical), which readily envelops the site seismic response spectra. The calculation assumes a maximum height of support to be 15 inches off of the wall/floor. All mounts using a smaller length of C-channel are qualified by comparison. The intermediate waveguide pipe supports and span lengths are qualified by a site specific calculation. There are two support designs used for mounting the waveguide pipe. A prequalified seismic support per BSEP specifications has been re-qualified for use with the increased BDB seismic loading of 2 times SSE. The second design uses a 3/4 inch thick steel base place with a four bolt anchorage pattern with a W-section column and cantilevered angle which uses a U-bolt configuration for attaching the 1 inch outside diameter pipe. The intermediate mounts use 2 times SSE seismic acceleration values at 20 Hz of 1.60g (horizontal) and 0.22g (vertical) per BSEP seismic design criteria specification.

All of the mounting supports for the waveguide piping are attached to either the Reactor Building concrete floor at elevation 117 feet 4 inch or to the Reactor Building north concrete wall. These concrete structures have a minimum concrete strength of 3000 pounds per square inch (psi). The conduit, local displays, remote displays, and Power Control Panels (PCP) are designed with the same 2 times SSE seismic criteria per each components' location within the Reactor Building or Control Building. Mounting calculations consider the total weight of the components and mounting brackets with additional allowances. The components are mounted with concrete anchor bolts to either Reactor Building or Control Building concrete walls rated for a minimum of 3000 psi concrete strength. The conduit along the overhead of the 50 foot elevation of the Reactor Building is mounted to existing plant steel girders. Conduit will be mounted using pre-qualified standard seismic supports per BSEP specifications that have been re-qualified to new BDB 2 times SSE seismic criteria.

The primary and back-up channel horn end mounting was evaluated per the electronic sensor and horn end assembly mount calculation which assumed a sloshing force value. Hydrodynamic loading to the horn assembly due to SFP sloshing from a seismic event has been evaluated per a site specific design calculation. The SFP sloshing loads were calculated for wave forces in the horizontal and vertical directions. The calculations concluded that the original assumed sloshing force envelops the actual calculated force.

The NRC staff reviewed the design criteria and methodology used to estimate and test the total loading on the mounting devices, including the design-basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing. Based on that review the staff finds that the loading on the mounting devices has been adequately determined. The staff also finds that the structural integrity of the affected structures was adequately addressed and that the mounting attachments were designed to a conservative seismic ground motion (2 times SSE).

Based on the evaluation above, the NRC staff finds that the licensee's SFP level instrument 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 is not already covered by existing quality assurance requirements. Per 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 augmented quality provisions will be applied to ensure the rigor of the qualification documentation reviews and in-plant modification installation oversight is sufficient to ensure compliance with the qualification requirements above.

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 Equipment Reliability

The NRC staff reviewed the AREVA SFPLI's qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic. The staff further reviewed the anticipated BSEP's environmental conditions during the on-site audit as follows.

4.2.4.2.1 Temperature, Humidity, and Radiation

By letter dated May 19, 2016 [Reference 21], the licensee stated that the postulated BDB temperature in the SFP is 100 degrees Centigrade (°C) (212°F) consistent with a boiling SFP. According to the licensee, the waveguide pipe, horn assembly, mounting supports, and horn cover located on the refuel floor adjacent to the SFP consist of passive metallic and glass components that are not adversely affected by this postulated temperature. In the area where the electronics are located, consisting of the sensor, indicator and PCP, the temperatures will not exceed the maximum rated temperatures of the electronics. The electronics in the sensor are rated for a maximum continuous duty temperature of 80°C (176°F) on the condition that the process temperature (that which the flange connection is in contact with) is no greater than 130°C (266°F). The sensors are located as far away from the SFP as practical, which places them within the stairwell on the 98' elevation. Per the licensee's environmental condition analysis, the maximum temperature at the 80-foot ceiling elevation (approximately 98-foot elevation) associated with the plant's coping and mitigation strategies is less than the rating of the electronic sensor.

The PCP internal components are rated for a maximum temperature of at least 70°C (158°F). Allowing for 5°C (9°F) heat rise in the panel, the overall panel maximum ambient temperature for operation is 65°C (149°F). The PCP enclosure is rated NEMA 4X and provides protection to the internal components from the effects of high humidity environments. The PCP and display is located within the control room in the back panel. These components along with the electronic sensor are the electronic components required for system function in a BDB. Per Environmental Qualification Service Conditions (EQSC) document DR-227 the control room is considered a mild environment in which at no time would it be subjected to an environment more severe as that which would occur during normal conditions, including anticipated operational occurrences. The normal temperature of the control room is 40°F to 120°F per this EQSC document. As described in Section 3.9.1.1 of this SE, the licensee's ventilation strategy will limit the control room temperature to a maximum of 116°F. The limiting component within the control room is the PCP which, as noted above, has a maximum operating temperature of 149°F.

Condensation formation on the inner waveguide pipe walls would require very moist air to enter the pipe at the sensor and travel to an area where the air temperature in the pipe would be at the dew point. This is a highly unlikely occurrence with the horn cover, which blocks airflow through the waveguide pipe. This reduces the potential for transfer of warm moist air to a colder area and therefore reduces the potential for condensation forming in the pipe.

By letter dated May 19, 2016 [Reference 21], the licensee stated that the area above and around the pool will be subject to large amounts of radiation in the event that the fuel becomes uncovered. The only parts of the measurement channel in the pool radiation environment are the metallic waveguide and horn, which are not susceptible to damage from the expected levels of radiation. The electronics must be located in an area that is shielded from the direct shine from the fuel, and bounce and scatter effects above the pool. For the purpose of the licensee's analysis, the radiation levels in the area are assumed to not exceed 1×10^3 rad (mild radiation environment). Dose rates used for testing electronics using MIL-STD-883J, Method 1019.9 are 50 rad per second or greater. A Total Integrated Dose (TID) calculation for BSEP has been performed which analyzes the BDBEE initiated dose and normal operational dose at the proposed location for the SFP level transmitters (electronic sensors). The resulting TID (event-initiated plus normal dose) is used to assess the viability of the transmitter location. A 5 year data set of radiological surveys was acquired for normal dose rates and used as a basis for projecting future dose accumulation. Based on the TID calculation with a limiting threshold of 1000 rads, the east side transmitter lifetime is computed to be approximately 53 years, while the west side transmitter lifetime is computed to be approximately 8 years. The difference in lifespan projections is a reflection of the normal dose rates surveyed in those areas, where the east side sensor location has very minimal normal dose and the west side is known to have two prominent point sources nearby. These sensor lifetimes are accounted for by periodic replacement to comply with two channel functionality. The VEGADIS 62 display and PCP electronic components located within the control room, which is a mild radiological environment.

The NRC staff review finds that the equipment qualifications envelop the anticipated BSEP radiation, temperature, and humidity conditions during a postulated BDBEE, and thus the SFPLI should maintain its functionality during the expected BDB conditions.

4.2.4.2.2 Shock and Vibration

By letter dated May 19, 2016 [Reference 21], the licensee stated that the VEGAPULS sensor and PLICSCOM indicating and adjustment module were tested in accordance with MIL-STD-901 D. The MIL-S-901 D test consisted of a total of nine shock blows, three through each of the three principal axes of the sensor, delivered to the anvil plate of the shock machine. The heights of hammer drop for the shock blows in each axis were one foot, three feet and five feet.

The VEGAPULS sensor has also been shock tested in accordance with EN60068-2-27, with ten shock blows applied along a radial line through the support flange. A VEGAPULS sensor and PLICSCOM module were successfully vibration tested in accordance with MIL-STD-167-1. The test frequencies ranged from 4 Hz to 50 Hz with amplitudes ranging from 0.048 inches at the low frequencies to 0.006 inches at the higher frequencies. The potential vibration environment around the spent fuel pool and surrounding building structure might contain higher frequencies than were achieved in the testing discussed above. Additional testing of the VEGAPULS 62 ER sensor was performed in accordance with EN 60068-2-6 Method 204 (except 4g, 200 Hz). This additional testing is considered to provide a stand-alone demonstration of the resistance to vibration of the VEGAPULS sensor and further substantiates the results of the MIL STD 167-1 testing.

The NRC staff reviewed the testing performed to ascertain the equipment reliability of the SFPLI with respect to shock and vibration. The staff finds that the test parameters envelop the BSEP's anticipated shock and vibration conditions during a postulated BDBEE.

4.2.4.2.3 Seismic

By letter dated May 19, 2016 [Reference 21], the licensee stated that the electronic sensor, PLICSCOM display, PCP, horn end of the waveguide, standard pool end and sensor mounting brackets, and waveguide pipe were successfully seismically tested in accordance with the requirements of IEEE 344-2004. The system was monitored for operability before and after the resonance search and seismic tests. The required response spectra used for the five Operating Basis Earthquakes (OBE) and one SSE in the test were taken from EPRI TR-107330 Figure 4-5. This test level exceeds the building response spectra for the locations where the equipment is installed. A hydraulic analysis was performed to calculate the forces caused by sloshing of the SFP inventory in response to a seismic event. The seismic spectrum for fuel pool sloshing was obtained by considering the ground motion response spectrum (GMRS) as submitted to the NRC under NTTF 2.1, Seismic. Adequate margin was ensured for this application for consideration of BDB conditions. Acceleration levels at 0.5 percent damping for SSE, applicable to the water in SFP were used for sloshing. The method of analysis is based on TID-7024, "Nuclear Reactors and Earthquakes, United States Atomic Energy Commission," dated August 1963, along with pool surface profile and fundamental mode. These are used to estimate the impacting fluid velocity and to calculate the horn drag and inertial forces based on a Morison-type equation. The existing plant structures for attachment of the SFPLI equipment are seismically qualified to the current design basis and can be considered seismically robust. All SFPLI equipment critical to the function of the system in response to a BDBEE are attached to either existing plant concrete walls/floors or plant structural steel. The additional load due to the weight of the system components has a negligible impact to the structural integrity of the robust existing plant structures. The supports attached to concrete floors or walls are designed using concrete strength of 3000 psi.

The NRC staff review notes that the SFPLI equipment was tested to the seismic conditions that envelop BSEP's expected highest SSE. The staff also concluded that the assumptions, analytical, and conclusion model used in the sloshing analysis for the radar horn, waveguide, and horn end assembly are adequate. Further SFP level instrument mounting qualifications are discussed in Section 4.2.3, "Design Features: Mounting," of this SE.

Based on the evaluation above, the NRC staff finds that the licensee's proposed instrument qualification process 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.

4.2.5 Design Features: Independence

By letter dated May 19, 2016 [Reference 21], the licensee stated that the design of the instrument channels for the SFP level indication is of the same technology, permanently installed, separated by distance, and electrically independent of one another. According to the licensee, the channels have their own sensing components separated in accordance with NEI

12-02, divisionally separate cable routes, and separate electronics. Instrument channels are each powered normally by a separate station power source. The potential for a common cause event to adversely affect both channels is prevented since each channel maintains the separation of Division I and II, physically and by power source. Each channel is a separate instrument loop system, from the waveguide horn to the level sensor to the two level indications. Control and power wiring for each channel are separated by different conduits and trays. There is no common housing for channel indication or power.

The NRC staff reviewed the justification provided in the licensee's letter, and verified the independence consistent with the provisions of NEI 12-02, Section 3.5, during a walk down conducted during the site audit. The staff notes that with the licensee's design for the SFP level instrument, the loss of one level measurement channel would not affect the operation of other independent channel under BDB event conditions. The instrument channels' physical separation is further discussed in Section 4.2.2 of this SE.

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 OIP, the licensee stated that instrument channels will each be powered normally by a separate station power source and will have rechargeable or replaceable batteries with sufficient capacity to maintain the level indication function until the normal power is restored.

By letter dated May 19, 2016 [Reference 21], the licensee further stated that the VEGAPULS 62 ER Through Air Radar system is designed to maintain its accuracy after a power interruption. If a loss of 120 Vac power occurs, the power control panel automatically switches over to battery backup, resulting in a momentary loss of power to the radar sensor. After a short period of time required to re-boot, the sensor will continue to perform its measurement function. The PCP contains eight Tadiran Model TL-5920 C-cell lithium batteries that provide backup power when normal 120 Vac power is not available.

By having separate power supplies, battery provisions, and the ability to resume monitoring after a power interruption without significantly affecting accuracy and reliability, the NRC staff finds 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 its OIP, the licensee stated that instrument channels will be designed such that they will maintain their design accuracy without recalibration following a power interruption or change in power source. Accuracy will consider SFP post-event conditions, e.g., saturated water or steam environment. Additionally, instrument accuracy will be sufficient to allow trained personnel to determine when the actual level exceeds the specified level of each indicating range (Levels 1, 2, and 3) without conflicting or ambiguous indication.

By letter dated May 19, 2016 [Reference 21], the licensee further stated that the estimated accuracy of the Through Air radar instrument including waveguide is ±1 inches at normal SFP conditions and ±3 inches at BDB conditions including radiation, temperature, humidity, post-seismic, and post-shock conditions that would be present if the SFP level were at the Level 2 and Level 3 datum points. The maximum allowed deviation from the instrument channel design accuracy that will be employed under normal operating conditions 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 is based upon the difference between readings from the two level instrument channels. The estimated design accuracy for each instrument is ±1 inches. The combined deviation between the two instruments after which calibration is needed is therefore ±2 inches based on a still water level in the pool.

According to AREVA, the manufacturer reference accuracy for the SFP level instrument is ±1 inch for normal conditions and ±3 inch for BDB conditions including 212°F saturated steam. The VEGAPULS 62 ER SFPI is designed to maintain its accuracy after a power interruption. AREVA performed testing on VEGAPULS 62 ER to verify channel accuracy after changing power sources without recalibration. Each instrument measured 10 different locations including the maximum and minimum locations of the instrument span. The instrument accuracy was within ±1 inch of the actual target location for all measurements taken. AREVA then tested the instrument accuracy of the two test units while performing a horn rotation test, and found the readings taken to be within ±1 in. Finally, separate steam and fluid drain tests were performed and the instrument readings stayed within ±3 in. of the original reading (under normal conditions).

The NRC staff review of the licensee's justification for this attribute concludes that accuracy of the instruments is sufficient to provide trained personnel with the necessary indication under both normal and BDB conditions. If implemented properly, the instrument channels should maintain the designed accuracy following a power source change or interruption without the need of recalibration.

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

By letter dated May 19, 2016 [Reference 21], the licensee stated that the installed equipment will be tested to validate functionality of an installed instrument channel within 60 days of a planned refueling outage considering normal testing scheduling allowances (e.g. 25 percent). Periodic testing requires the measurement of two different points for validating accuracy and performing any adjustments to bring accuracy to within the required tolerances. In accordance with the vendor operating manual, the two points of measurement must be a minimum of 2 inches apart. In order to obtain two points of measurement, the actual water level will be used for one point and a metal calibration target will be used to obtain a simulated second point. Calibration procedure 0LP-LT008, "VEGAPULS 62 Level Transmitter Calibration" has been prepared for calibration of the SFPLI Radar Measurement System. The SFPLI scaling is from the top of the fuel racks or 16'-8" (0 percent indicated level) pool level up to a pool level of 37'-11" (100 percent indicated level). A calibration tolerance of ±1 inch was used as the acceptance criteria for the factory acceptance testing. This tolerance was also used for the site acceptance testing and is used when performing the periodic instrument calibrations prior to planned refueling cycles, as discussed above.

Procedure 0LP-LT008 provides the instructions for testing and calibrating the SFPLI equipment. This procedure performs removal from service, backup battery test, calibration check, calibration, return to service, and post maintenance activities. Within this procedure, the SFPLI indication levels are verified against the installed SFP ruler gauge level and against each other to ensure level instrumentation accuracy. Level indication for channels "A" and "B" shall be within ±1 inch of the ruler gauge level and the Channel "A" & "B" indications shall be within ±1 inch of each other.

Weekly channel checks will be performed using plant procedures to ensure the Channel "A" and Channel "B" local and remote indications are aligned. Reactor operator daily checks, performed once per 7 days, compare new control room SFP level indicator readings to each other (i.e., ±2 inch). Reactor Building auxiliary operator daily checks, performed once per 7 days, compare new Reactor Building SFP level indicator readings to each other (i.e., ±2 inch).

The NRC staff review finds that the SFPLI is adequately designed to provide the capability for routine testing and calibration including in-situ testing/calibration. By comparing the levels in the instrument channels and the maximum level allowed deviation for the instrument channel design accuracy, the operators could determine if recalibration or troubleshooting is needed.

Based on the evaluation above, the NRC staff finds that the licensee's proposed SFP instrumentation design allows for testing 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.

4.2.9 Design Features: Display

By letter dated May 19, 2016 [Reference 21], the licensee stated that displays for the primary and back-up channels are mounted on the north wall (Unit 1) and south wall (Unit 2) of the control room back panel area.

The NRC staff review finds that if implemented properly, the displays will provide continuous indication of SFP water level. The displays are located in seismically qualified buildings and the accessibility of the control room following an ELAP event is considered acceptable. During the onsite audit, the staff walked down and verified that the SFPI display locations should be promptly accessible and should remain habitable.

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 SFP 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 personnel shall be trained in the use and the provision of alternate power to each instrument channel. Station personnel performing functions associated with the SFPLI will be trained to perform the job specific functions necessary for their assigned tasks. 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.

The NRC staff finds that the use of SAT to identify the training population and to determine both the elements of the required training is acceptable. The licensee's proposed plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFPI and the provision of alternate power to the primary and backup instrument channels, 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 OIP, the licensee stated that procedures will be developed using guidelines and vendor instructions to address the maintenance, operation and abnormal response issues associated with the SFPLI. Procedures will also address strategy to ensure SFP water addition is initiated at an appropriate time consistent with implementation of NEI 12-06.

By letter dated May 19, 2016 [Reference 21], the licensee provided a description of BSEP procedures related to the SFPLI. Operations surveillance items include weekly channel checks on both the local and control room indications for both channels every 7 days. The maintenance procedures perform periodic calibration of the primary and back-up instruments, periodic replacement of the level sensors, periodic replacement of the PCP batteries, and periodic calibration of the AREVA radar measurement system.

The NRC staff reviewed selected licensee procedures that have been established for the testing, surveillance, calibration, and operation of the primary and backup SFP level instrument channels. The staff finds that the licensee's procedure development appears to be consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

By letter dated May 19, 2016 [Reference 21], the licensee provided a description of plant procedure 0PLP-01.4, "Fukushima FLEX System Availability, Action and Surveillance Requirements," which includes controls for the SFPI. This procedure specifies the required frequency for performance of periodic channel checks and functional checks, as appropriate. The procedure outlines allowed out of service time frames and specifies required remedial actions, in the event one or more channels cannot be restored within the allowed out of service time-frame. The procedure further requires functional testing be performed to verify proper channel operation within 60 days of a planned refueling outage, as specified in NEI 12-02. According to the licensee, the maintenance of the instruments is controlled by the plant preventive maintenance program and uses a combination of monitoring, predictive, preventative and replacement actions to ensure reliable equipment operation. The corrective action program

(CAP) would formally evaluate "functionality" for the SFP level channels and establish appropriate compensatory measures. The CAP would further establish appropriate procedural and process controls to ensure performance of any required compensatory measures.

As specified in NEI 12-02, compensatory actions must be implemented if one channel is not expected to be restored to functional within 90 days. Procedure 0PLP-01.4 lists the following potential actions for a channel that cannot be restored to functional status within 90 days:

- More frequent surveillance (channel check) to verify functionality of the remaining level channel or strategy
- Implementation of equipment protective measures
- Increased operator visual surveillance of the SFP level and area
- Maintenance of elevated SFP level
- Reduction of SFP temperature
- Supplemental operations staffing

As specified in NEI 12-02, if both the primary and backup systems are out of service within 24 hours action to restore one channel to service must be initiated and compensatory actions must be implemented with 72 hours. As specified in procedure 0PLP-01.4, the licensee's compensatory actions could include:

- Increased operator visual surveillance of the SFP level and area
- Maintenance of elevated SFP level
- Reduction of SFP temperature
- Supplemental operations staffing
- Pre-staging of FLEX support equipment (nozzles, hoses, etc.) which are relied upon for SFP makeup (pre-staged equipment would be located within seismic category I structures)

The NRC staff review notes that the testing and calibration described in the licensee's compliance letter are consistent with the vendor recommendations. Additionally, the compensatory actions for instrument channel(s) out-of-service appear to be consistent with the guidance in 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

By letter dated May 19, 2016 [Reference 21], 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 SFPLI is installed at BSEP, Units 1 and 2, 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 December 2013 [Reference 19]. The licensee reached its final compliance date on March 23, 2016, and has declared that both of the reactors are in compliance with the orders [References 20 and 21]. The purpose of this SE 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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2. 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," February 17, 2012 (ADAMS Accession No. ML12039A103)

3. SRM-SECY-12-0025, "Staff Requirements – 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," March 9, 2012 (ADAMS Accession No. ML120690347)

4. Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," March 12, 2012 (ADAMS Accession No. ML12054A736)

5. Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," March 12, 2012 (ADAMS Accession No. ML12054A679)

6. Nuclear Energy Institute document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0, August 21, 2012 (ADAMS Accession No. ML12242A378)

7. JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," August 29, 2012 (ADAMS Accession No. ML12229A174)

8. Nuclear Energy Institute 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, dated August 24, 2012 (ADAMS Accession No. ML12240A307)

9. JLD-ISG-2012-03, "Compliance with Order EA-12-051, Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," August 29, 2012 (ADAMS Accession No. ML12221A339)

10. Letter, Duke to NRC, "Brunswick, Units 1 and 2 - Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 28, 2013 (ADAMS Accession No. ML13071A559)

11. Letter, Duke to NRC, "Brunswick, Units 1 and 2 First Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated August 20, 2013 (ADAMS Accession No. ML13248A447)

12. Letter, Duke to NRC, "Second Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 28, 2014 (ADAMS Accession No. ML14073A451)

13. Letter, Duke to NRC, "Brunswick, Units 1 and 2 - Third Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated August 28, 2014 (ADAMS Accession No.r ML14254A176)

14. Letter, Duke to NRC, "Brunswick, Units 1 & 2, Fourth Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 27, 2015 (ADAMS Accession No. ML15084A156)

15. Letter, Duke to NRC, "Brunswick, Units 1 and 2 - Fifth Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated August 26, 2015 (ADAMS Accession No. ML15246A034)

16. Letter, Duke to NRC, "Brunswick Steam Electric Plant - Sixth Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 24, 2016 (ADAMS Accession No. ML16074A394)

17. Letter from Jack R. Davis (NRC) to All Operating Reactor Licensees and Holders of Construction Permits, "Nuclear Regulatory Commission Audits of Licensee Responses to Mitigation Strategies Order EA-12-049," August 28, 2013 (ADAMS Accession No. ML13234A503)

18. Letter from Jeremy S. Bowen (NRC) to George T. Hamrick, "Brunswick Steam Electric Plant, Units 1 and 2 - Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigation Strategies) (TAC Nos. MF0975 and MF0976)," dated November 22, 2013 (ADAMS Accession No. ML13220A090)

19. Letter from Peter Bamford (NRC) to William R. Gideon, "Brunswick Steam Electric Plant, Units 1 and 2 - Report for the Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051 (TAC Nos. MF0975, MF0976, MF0973 and MF0974)," dated March 31, 2015 (ADAMS Accession No. ML15082A155)

20. Letter, Duke Energy to NRC, "Brunswick, Units 1 and 2- Notification of Full Compliance with 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" for BSEP, Unit 2," dated June 3, 2015 (ADAMS Accession No. ML15173A013)

21. Letter, Duke to NRC, "Brunswick Steam Electric Plant - Notification of Full Compliance with 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" for BSEP, Units 1 and 2," dated May 19, 2016 (ADAMS Accession No. ML16146A604)

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24. Letter from Jack R. Davis (NRC) to Joseph E. Pollock (NEI), "Staff Assessment of National SAFER Response Centers Established In Response to Order EA-12-049," September 26, 2014 (ADAMS Accession No. ML14265A107)

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26. Letter from Christopher Gratton (NRC) to Michael J. Annacone, "Brunswick Steam Electric Plant, Units 1 and 2 – Request for Additional Information Regarding Overall Integrated Plan in Response to Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," May 23, 2013 (ADAMS Accession No. ML13141A622)

27. Letter, Duke to NRC, Brunswick Steam Electric Plant, Units 1 and 2, "Response to Request for Additional Information Regarding Overall Integrated Plan in Response to Order EA-12-051, 'Reliable Spent Fuel Pool Instrumentation'," July 22, 2013 (ADAMS Accession No. ML13219B117)

28. Letter from Christopher Gratton (NRC) to George T. Hamrick, "Brunswick Steam Electric Plant, Units 1 and 2 – Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan For Implementation of Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," November 18, 2013 (ADAMS Accession No. ML13269A345)

29. Letter, Duke to NRC, "Duke Energy Progress, Inc., First Six-Month Status Report in Response to March 12, 2012, Commission Order with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 26, 2013 (ADAMS Accession No. ML13242A010)

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33. Letter, Duke to NRC, "Brunswick, Units 1 and 2 - Fifth Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 26, 2015 (ADAMS Accession No. ML15246A033)

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36. NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, Regulatory Audits, December 16, 2008 (ADAMS Accession No. ML082900195)

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38. NEI Position Paper: "Shutdown/Refueling Modes," dated September 18, 2013 (ADAMS Accession No. ML13273A514)

39. Letter from Jack R. Davis (NRC) to Joseph E. Pollock (NEI), regarding NRC endorsement of NEI Position Paper: "Shutdown/Refueling Modes," dated September 30, 2013 (ADAMS Accession No. ML13267A382)

40. Letter from Nicholas Pappas (NEI) to Jack R. Davis (NRC) regarding FLEX Equipment Maintenance and Testing dated October 3, 2013 (ADAMS Accession No. ML13276A573)

41. Letter from Jack R. Davis (NRC) to Joseph E. Pollock (NEI), regarding NRC endorsement of the use of the EPRI FLEX equipment maintenance report, dated October 7, 2013 (ADAMS Accession No. ML13276A224)

42. Letter from Jack R. Davis (NRC) to Joseph E. Pollock (NEI), regarding MAAP use in support of post-Fukushima applications, dated October 3, 2013 (ADAMS Accession No. ML13275A318)

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46. U.S. Nuclear Regulatory Commission, "Mitigation of Beyond-Design-Basis Events," *Federal Register*, Vol. 80, No. 219, November 13, 2015, pp. 70610-70647

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Date: December 14, 2016

These reports were required by the order, and are listed in the attached safety evaluation. By letters dated November 18, 2013 (ADAMS Accession No. ML13269A345), and March 31, 2015 (ADAMS Accession No. ML15082A155), 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 June 3, 2015 (ADAMS Accession No. ML15173A013), Duke reported that BSEP, Unit 2 was in full compliance with Order EA-12-051. By letter dated May 19, 2016 (ADAMS Accession No. ML16146A604), Duke reported that BSEP, Unit 1 was in full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Duke's strategies for BSEP, Units 1 and 2. The intent of the safety evaluation is to inform Duke 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 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML14273A444). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact Peter Bamford, Orders Management Branch, BSEP Project Manager, at 301-415-2833 or at Peter.Bamford@nrc.gov.

Sincerely,
/RA/
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Docket Nos.: 50-325 and 50-324
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