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

August 4, 2017

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295 / BIN B038
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR 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. MF0712, MF0713, MF0721, AND MF0722)

Dear Mr. Hutto:

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 27, 2013 (ADAMS Accession No. ML13059A385), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted its OIP for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Hatch), in response to Order EA-12-049. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 27, 2014 (ADAMS Accession No. ML13364A202), and January 13, 2016 (ADAMS Accession No. ML15349A801), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated February 13, 2017 (ADAMS Accession No. ML17045A597), SNC submitted a compliance letter and Final Integrated Plan (FIP) in response to Order EA-12-049. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-049.

By letter dated February 27, 2013 (ADAMS Accession No. ML13059A389), the licensee submitted its OIP for Hatch 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 October 28, 2013 (ADAMS Accession No. ML13294A529), and January 13, 2016 (ADAMS Accession No. ML15349A801), 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 February 13, 2017 (ADAMS Accession No. ML17044A414), SNC submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

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

If you have any questions, please contact Jason Paige, Orders Management Branch, Hatch Project Manager, at 301-415-1474 or at Jason.Paige@nrc.gov.

Sincerely,

Tony Brown, Acting Chief
Orders Management Branch
Japan Lessons-Learned Division
Office of Nuclear Reactor Regulation

Docket Nos.: 50-321 and 50-366

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

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" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12054A736). 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" (ADAMS Accession No. ML12054A679). This order directed licensees to install reliable SFP level instrumentation with a primary channel and a backup channel, and with independent power supplies that are independent of the plant alternating current (ac) and direct current (dc) power distribution systems. Order EA-12-051 applies to all power reactor licensees and all holders of construction permits for power reactors.

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC 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 (ADAMS Accession No. ML11186A950). 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" (ADAMS Accession No. ML12039A103), 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 (ADAMS Accession No. ML120690347), the NRC staff issued Orders EA-12-049 and EA-12-051.

2.1 Order EA-12-049

Order EA-12-049, Attachment 2 (ADAMS Accession No. ML12054A736), 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 December 10, 2015, following submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, Revision 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (ADAMS Accession No.

ML16005A625), to the NRC to provide revised specifications for an industry-developed methodology for the development, implementation, and maintenance of guidance and strategies in response to the Mitigation Strategies order. The NRC staff reviewed NEI 12-06, Revision 2, and on January 22, 2016, issued Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 1, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (ADAMS Accession No. ML15357A163), endorsing NEI 12-06, Revision 2, with exceptions, additions, and clarifications, as an acceptable means of meeting the requirements of Order EA-12-049, and published a notice of its availability in the *Federal Register* (81 FR 10283).

2.2 <u>Order EA-12-051</u>

Order EA-12-051, Attachment 2 (ADAMS Accession No. ML12054A679), requires that operating power reactor licensees and construction permit holders install reliable SFP level instrumentation. Specific requirements of the order are listed below:

> All licensees identified in Attachment 1 to the order shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.
>
> 1. The spent fuel pool level instrumentation shall include the following design features:
>
> 1.1 Instruments: The instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.
>
> 1.2 Arrangement: The spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.
>
> 1.3 Mounting: Installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

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

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

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

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

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

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

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

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

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

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

On August 24, 2012, following several NEI submittals and discussions in public meetings with NRC staff, the NEI submitted document NEI 12-02, "Industry Guidance for Compliance With NRC Order EA-12-051, To Modify Licenses With Regard to Reliable Spent Fuel Pool

Instrumentation," Revision 1 (ADAMS Accession No. ML12240A307) 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" (ADAMS Accession No. ML12221A339), 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 27, 2013 (ADAMS Accession No. ML13059A385), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted its Overall Integrated Plan (OIP) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Hatch), in response to Order EA-12-049. By letters dated August 27, 2013 (ADAMS Accession No. ML13240A238), February 26, 2014 (ADAMS Accession No. ML14057A776), August 26, 2014 (ADAMS Accession No. ML14239A650), February 26, 2015 (ADAMS Accession No. ML15057A315), August 27, 2015 (ADAMS Accession No. ML15239B301), February 25, 2016 (ADAMS Accession No. ML16056A632), and August 8, 2016 (ADAMS Accession No. ML16221A397), the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 27, 2014 (ADAMS Accession No. ML13364A202), and January 13, 2016 (ADAMS Accession No. ML15349A801), the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated February 13, 2017 (ADAMS Accession No. ML17045A597), the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP).

3.1 Overall Mitigation Strategy

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

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

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

Hatch is a General Electric boiling-water reactor (BWR) Model 4 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 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. Because the upper portion of the condensate storage tank (CST) is not robust against wind-driven missiles, the licensee's mitigating strategy assumes that the RCIC pump suction realigns to the suppression pool after the CST is depleted. Within 3 hours and 19 minutes, the operators take manual control of the SRVs to perform a controlled cooldown and depressurization of the reactor. The initial cooldown of the primary system is stopped when reactor pressure reaches a control band of 380 pounds per square inch gauge (psig) to 480 psig to ensure sufficient steam pressure to operate the RCIC pump. The SRV blowdown and RCIC pump exhaust will cause the suppression pool temperature to rise. Therefore, when the suppression pool heats up to a predetermined setpoint, the hardened containment vent system (HCVS) is opened to mitigate the temperature rise and extend operation of the RCIC system. The RPV makeup will continue to be provided from the RCIC system, but as the CST is depleted the suction source of the RCIC pump will be swapped to the suppression pool. Transition to Phase 2 is considered when the RCIC pump is realigned to take suction from the refilled CST.

The primary method for maintaining core cooling during Phase 2 will be by utilizing the RCIC system. As stated above, the suction for the RCIC system will be swapped back to the refilled CST from the suppression pool. The CST will be refilled by using the FLEX pump which will take suction from the Altamaha River using two submersible booster pumps with integral strainers. As an alternate strategy, the FLEX pump can also be used to provide direct injection to the RPV via the residual heat removal service water (RHRSW) system to the RHR system.

Both reactors have Mark I containments which are inerted with nitrogen at power. The licensee performed a containment evaluation and determined that opening the suppression pool vent to atmosphere will maintain containment temperature and pressure within acceptable levels until equipment from the National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) can be set up for cooling of the suppression pool.

Each reactor has a SFP in its reactor building. To maintain SFP cooling capabilities, the licensee stated that the required action is to establish ventilation and 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 SFP will initially heat up due to the unavailability of the normal cooling system. The licensee has calculated that, depending on the heat loading in the pool, boiling could start as soon as 4.22 hours (full-core offload) after the start of the ELAP. However, during normal operation, the SFP water inventory will heat up from 110°F to 212°F during the first 12 hours of an ELAP. The pool water level would drop to 15 feet (ft.) above the top of the fuel racks in approximately 18 hours (full-core offload). The licensee determined that habitability on the pool operating deck area could become compromised as early as 12 hours (design-basis maximum heat load) after the ELAP, so valve lineups and hose deployments are planned prior to that time.

To makeup to the SFP, the licensee has a primary and alternate strategy to account for the condition of the pool. If the refuel floor is accessible and habitable, the primary SFP strategy is to connect FLEX hoses to a diesel-powered FLEX pump. The discharge ends of these hoses are either routed all the way to the SFPs with the discharge ends positioned over the edge of the pool or connected to nozzles that spray the water over the spent fuel. If the refuel floor is not accessible or habitable, the alternate strategy is to provide makeup to the SFP by connecting a FLEX hose from a diesel-powered FLEX pump via the RHRSW system piping at the intake structure to the plant service water (PSW) system crosstie. The PSW piping provides an emergency fill connection to the SFP cooling system makeup piping.

The operators will perform dc bus load stripping within the initial 75 minutes following event initiation to ensure safety-related battery life is extended greater than 14 hours. Following dc load stripping and prior to battery depletion, one 545-kilowatt (kW), 600 volt alternating current (Vac) generator will be deployed from a FLEX storage building (FSB) to each unit. These portable generators will be used to repower essential battery chargers within 10 hours of ELAP initiation.

For Phase 3, a NSRC will provide high capacity pumps and large combustion turbine-driven generators which could be used to restore one residual heat removal (RHR) cooling train per unit to cool the cores in the long-term. There are two NSRCs in the United States.

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

3.2 Reactor Core Cooling Strategies

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

As reviewed in this section, the licensee's core cooling analysis 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. 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 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

In its FIP, the licensee stated that the injection of water into the RPV will be accomplished through the RCIC system. The RCIC system suction is initially lined up to the CST and RCIC will pump water into the core from the CST automatically. The CST is seismically robust, but the upper portion is vulnerable to wind-driven missiles. Although the CST has a capacity of 500,000 gallons, only 100,000 gallons are protected against wind hazards and are credited to be available in all FLEX scenarios. The high pressure coolant injection (HPCI) system will also automatically start with suction aligned to the CST and discharge to the RPV.

Both the RCIC and HPCI pumps are powered by turbines using steam from the RPV and are robust for the hazards considered in the ELAP evaluation. Both the RCIC and HPCI pumps are designed to automatically start following the ELAP event. Following the initial restoration of RPV water level, operators will secure the HPCI system at approximately 4 minutes after the initiation of the ELAP event. In the event that RCIC does not automatically start, procedural guidance directs the operators to manually start the pump. RCIC discharges into the RPV feedwater line. The RCIC system valves are powered by the 125 volt direct current (Vdc) bus and are used to control the cooling flow to the RPV, balancing it with the outflow of steam through the SRVs to the suppression pool in order to maintain the RPV level within its desired control band.

Pressure control of the RPV is accomplished using the SRVs, which are powered by the station batteries. The SRV's normal pneumatic supply is lost at event initiation due to the loss of power to the air compressors. Each of the SRV's is equipped with backup accumulators for SRV operation. The SRV's will additionally be supplied with pneumatic pressure maintained by a liquid nitrogen storage tank that will automatically provide a backup supply. The unit specific nitrogen storage tanks can be cross-tied to supply a large pneumatic storage volume. During Phase 1, the SRV's will be used to lower RPV pressure to an initial pressure range of 380 to 480 psig at a cooldown rate of approximately 100 degrees Fahrenheit (°F) per hour.

After the start of the ELAP, SRV blowdown and RCIC pump exhaust will cause the suppression pool temperature to rise. The licensee will use the HCVS to maintain suppression pool temperature below 230°F. This venting will stop the rise of the suppression pool temperature, but will also lower the suppression pool inventory. The licensee expects to open the vent at approximately 5 hours after the initiation of the ELAP event. The vent system is provided with power from the dc bus systems. The pneumatically operated valves in the vent system have two 400 gallon accumulators, as well as backup nitrogen bottles to provide motive force for operations. The HCVS is robust for all applicable hazards as defined in NEI 12-06.

The SRVs are safety-related equipment that are located inside the plant drywell and protected from all external hazards as defined in NEI 12-06. The backup nitrogen system is also seismically qualified and will not isolate from the SRV's on a loss of power. The SRV's are powered from the Class 1E 125V dc distribution system. In its FIP, the licensee stated that the SRV solenoids have been environmentally tested at elevated temperatures and pressures. The test conditions were in excess of the conditions that the Modular Accident Analysis Program (MAAP) analysis indicates that the drywell will reach during the ELAP event (a steady state temperature of 230°F).

Hatch's batteries and Class 1E 125 Vdc distribution system provide power to RCIC system, SRVs and required instrumentation. A dc load shed is completed approximately 75 minutes after the initiation of the ELAP event to extend the battery capacity to power the Phase I systems and instruments. Installed batteries can maintain necessary voltage for at least 14 hours. Prior to battery depletion, a 600 Vac FLEX diesel generator (DG) is deployed and used to recharge the station batteries.

The licensee intends to transfer the RCIC system suction to the suppression pool when the CST is nearing depletion. The credited 100,000 gallons in the CST will provide for approximately 6.8 hours of makeup water to the RPV. The suppression pool temperature is expected to be approximately 215°F at this point.

3.2.1.2 Phase 2

Per the FIP, Hatch's primary method of maintaining core cooling during Phase 2 will be by utilizing the RCIC system. The suction for the RCIC system will be swapped from the suppression pool back to the CST, after the CST is refilled by a FLEX pump taking suction from the Altamaha River.

As an alternate strategy to using the RCIC pump, the FLEX pump can also provide direct injection to the RPV. The FLEX pump is rated for 3,000 gallons per minute (gpm) at 150 psi. The FLEX pump will take suction from the Altamaha River using two submersible booster pumps with integral strainers and discharge to one of two connections. The primary connection is a FLEX connection located on the RHRSW ground level that will discharge to the RPV via either division of RHR injection flow path. The alternate connection point is a FLEX connection on the RHRSW line in the reactor building. There are two FLEX pumps stored onsite to provide N+1.

In order to maintain the continued use of the SRVs, RCIC and required instrumentation, a 600 Vac FLEX generator will be deployed to supply power to the plant battery chargers at approximately 10 hours. The plant batteries are sized to power the dc systems for approximately 14 hours without recharging.

3.2.1.3 Phase 3

The Phase 3 strategy includes the use of equipment from the NSRC. The plant plans to continue the use of Phase 2 equipment or replace it with SAFER equipment as necessary. Water will be injected into the RPV from the RCIC system as long as sufficient steam pressure is available. Water will continue to be supplied from the Altamaha River. The NSRC supplied equipment will be used to provide makeup to the CST for indefinite core cooling. The NSRC supplied equipment will include a medium flow pump, a high flow pump, an RPV makeup pump, a high pressure injection pump, a water treatment system, a 4160 V generator and a 4160 V distribution system, a 480 V generator and a 480 to 600 V transformer.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

In its FIP, the licensee stated that the Hatch site elevation is above the maximum plant site flood level. The grade elevation of all safety-related structures is approximately 129 ft. mean sea level (MSL) and the grade of the intake structure is 110 ft. MSL. The maximum probable flood elevation without wave runup is 105 ft. MSL and with wave runup the water level may reach 108.3 ft. MSL. In its FIP, the licensee stated that the FSB is located above the upper-bound flood elevation and the deployment path of FLEX pumps to the UHS remains accessible. Additionally, there is no large enclosed, or partially enclosed, body of water adjacent to the site. Therefore, no variations to the core strategy are necessary at this time.

3.2.3 Staff Evaluations

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

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

3.2.3.1.1 Plant SSCs

The licensee described in its FIP that the RCIC system is used to provide makeup water to the RPV following an ELAP event. The RCIC system consists of a steam-driven turbine-pump, associated piping, and discharge motor-operated isolation/flow control valves that are used to modulate the flow to maintain the required RPV water level. The RCIC pump can take suction from the CST or from the suppression pool. The CST is used as the initial source of water for RPV make-up and is the source for makeup water after RCIC suction is transferred back to the CST from the suppression pool. Both the CST and suppression pool are described in more detail below in Section 3.10 of this safety evaluation. The motor operated valves (MOVs) are supplied with dc power from station batteries and later the 600 V FLEX DG. The valves can be controlled remotely from the control room or operated manually if dc power is lost. The RCIC pump, piping, and discharge motor-operated isolation/flow control valves are safety-related, seismically qualified components located in the reactor building. The reactor building is a seismically qualified structure and is protected from all applicable external hazards.

The licensee also described in the FIP that the SRVs release the steam from the reactor at the initial portion of the ELAP event. The nitrogen storage tanks and nitrogen system piping are used to supply nitrogen to the SRVs to function during the ELAP event. The nitrogen supply system piping from the nitrogen storage tanks has the ability to be cross-connected between Unit 1 and Unit 2 such that if one of the tanks should be damaged or have insufficient nitrogen pressure for SRV operation, the other tank can supply the necessary nitrogen for SRV operation. As backup to the existing nitrogen tank and the SRV accumulators, pre-staged emergency nitrogen bottles can be used as directed in existing site procedures. The SRVs, nitrogen storage tanks, and associated piping are seismically protected components located within the reactor building.

Based on the location and design of the credited plant SSCs, as described in the licensee's FIP and the plant's updated final safety analysis report (UFSAR), the credited plant SSCs should be available to support core cooling during an ELAP, consistent with NEI 12-06, Section 3.2.1.3.

3.2.3.1.2 Plant Instrumentation

In its FIP, the licensee stated that the Hatch's plan is to monitor instrumentation in the control room and by alternate means if necessary to support the FLEX cooling strategy. The instrumentation is powered by the station batteries and will be maintained available for indefinite coping via battery chargers powered by the FLEX DGs. A more detailed evaluation of the instrumentation power supply is contained in Section 3.2.3.6 of this safety evaluation.

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

- Reactor water level
- Suppression pool water level
- Drywell and suppression pool temperature
- Drywell and suppression pool pressure
- Condensate storage tank level
- Dc bus voltage

These instruments are monitored from the control room and are accessible to the operators throughout the event. The instrumentation identified by the licensee to support its core cooling strategy is consistent with the recommendations specified in the endorsed guidance of NEI 12-06.

Per the FIP, the station batteries normally power the instrumentation during an ELAP event. Following the initiation of the ELAP event, load shedding of the station batteries will be completed within 75 minutes extending the battery life up to at least 14 hours, to maintain the availability of critical instruments and controls. Charging of the batteries will be initiated within 10 hours of the ELAP event by means of a FLEX portable 600 Vac DG. The FLEX generator will continue to provide power through the duration of the event. Additional backup generators will be available from the NSRC during Phase 3. Therefore, based upon the information provided by the licensee, the NRC staff understands that the critical instruments will be accessible continuously throughout the ELAP event. In addition, the FIP stated that the portable FLEX equipment will have local instrumentation necessary for operating the equipment.

In accordance with NEI 12-06 Section 5.3.3.1, guidelines for obtaining critical parameters locally are provided to the operators. The FIP stated that contingencies for alternate instrumentation monitoring are provided for the operators following a BDBEE for essential instrumentation.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee based its mitigating strategy for reactor core cooling in part on thermal-hydraulic analysis performed using Version 4 of MAAP4. 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 safety evaluation 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 safety evaluation.

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

To provide analytical justification for their mitigating strategies in response to Order EA-12-049, a number of licensees for BWRs and pressurized-water reactors (PWRs) completed analyses of the ELAP event using 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 analysis with the TRACE code to obtain an independent assessment of the predictions of the MAAP4 code.

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

Based on the NRC staff's review of EPRI's June 2013 technical report, as supplemented by further discussion with the code vendor, audit review of key sections of the MAAP code documentation, and confirmation of acceptable agreement with NRC staff simulations using the TRACE code, the NRC staff concluded that, under certain conditions, the MAAP4 code may be used for best-estimate prediction of the ELAP event sequence for BWRs.

The NRC staff issued an endorsement letter dated October 3, 2013, which documented these conclusions and identified specific limitations that BWR licensees should address to justify the applicability of simulations using the MAAP4 code for demonstrating that the requirements of Order EA-12-049 have been satisfied.

During the 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 a review of the MAAP4 calculation documentation, the NRC 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 technical specification limits, were satisfied. Specifically, the licensee's analysis calculated that Hatch 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 8.75 ft. 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, Hatch'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.

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 MAAP4 calculations for Hatch assumed a total leakage rate at normal RPV operating pressure of 36 gpm. This leakage rate includes 18 gpm per recirculation pump seal, which is in accordance with work performed to support NRC Generic Letter 91-07.

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 the FIP, the licensee stated that the seal leakage rate and total RCS leakage rate will be proportional to the RPV pressure. Based on the licensee's analysis, the FLEX pump is sized to meet the decay heat removal requirements at one hour after reactor trip. In this condition the decay heat load would require a total flow rate of 685 gpm for each unit at 512 ft. of total head. The FLEX pump is able to provide 1500 gpm at 575 ft. of discharge head. The pump is deployed at approximately 10 hours after event initiation. 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 per pump is reasonable. Gross seal failures are not anticipated to occur during the postulated ELAP event. As is typical of the majority of U.S. BWRs, Hatch 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 on 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 Hatch's UFSAR, the control rods provide adequate shutdown margin under all anticipated plant conditions, with the assumption that the highest-worth control rod remains fully withdrawn. Hatch technical specification 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 is conservative 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 and Water Supplies

In its FIP, the licensee stated that the FLEX pump will be available during Phase 2 to take suction from the Altamaha River 10 hours into the ELAP event to discharge makeup water into the RPV and/or the CST through the RHRSW piping in the intake structure, which leads into the reactor building. The FLEX pump is a centrifugal diesel-driven pump, and contains four suction and four discharge hose connections. The FLEX pump is also accompanied by two submersible, hydraulically driven pumps with floatation and suction mesh screens for filtration of the water suctioned from the Altamaha River. Additional portable mesh strainers are available for the PSW system and RHRSW crosstie for further filtration of the water going to the control room ventilation coolers. The licensee's vendor document A-47460, "FLEX Portable System – FLEX Pump Vendor Manual," Revision 0, indicates that the FLEX pump is capable of providing 3000 gpm at 150 psig, which exceeds the required flow rate for the bounding case. The licensee determined that the required flow rate is based on the bounding case, which calls for the FLEX pump to discharge water from the Altamaha River to the RHRSW piping and supply makeup water to both Units 1 and 2 for RPV makeup via CST makeup and the RCIC system, SFP makeup via the new FLEX connection in the reactor building, control room cooling, and RCIC room cooling. The licensee stated that the makeup strategies can be executed with one FLEX pump, with the additional FLEX pump serving as "N+1." Both of the FLEX pumps will be stored in the FSB, which is protected from all applicable external hazards.

The FLEX strategies also use a diesel-driven FLEX air compressor to supply compressed air to operate a pneumatic wrench for RHR valve manipulation and to maintain the SFP transfer canal gap seals inflated. Three FLEX air compressors are available, one for each unit and the third as "N+1." The air compressors are also stored in the FSB.

During the audit, the NRC staff reviewed the flow rates and pressures evaluated in the hydraulic analyses, as summarized above. The NRC staff notes that the performance criteria for the Phase 3 NSRC equipment are consistent with the capacities of the Phase 2 FLEX equipment. Based on the NRC staff's review of the FLEX equipment at Hatch, as described in the above hydraulic analyses and the FIP, the NRC staff concludes that the FLEX equipment should perform as intended to support core cooling and RCS inventory control 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 an ELAP and LUHS. The electrical strategies described in the licensee's 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 safety evaluation.

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

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

During the first phase of the ELAP event, Hatch would rely on the Class 1E station batteries to provide power to key instrumentation and power to controls for SSCs used to maintain the key safety functions (reactor core cooling, RPV inventory control, and containment integrity). The Hatch Class 1E station batteries and associated dc distribution systems are located in a safety-related structure designed to meet all applicable external hazards. The licensee's procedures 31EO-FSG-003-1, "FSG - ELAP (Extended Loss of AC Power)," Version 1.2, and 31EO-FSG-003-2, "FSG - ELAP (Extended Loss Of AC Power," Version 2.0, direct operators to conserve dc power during the event by stripping non-essential loads. This will extend battery life until backup power (Phase 2) is available. The plant operators would commence load shedding no later than 48 minutes and complete load shedding within 75 minutes from the onset of an ELAP event.

The station 125/250 Vdc system for each unit consists of two 125/250 Vdc independent and redundant switchgear assemblies, six static-type battery chargers, and two independent service batteries A and B. The station 125/250 Vdc system provides power to critical instruments, SRVs, emergency control room lighting, RCIC system, and other required dc loads.

The Class 1E station batteries were manufactured by C&D Technologies. Class 1E station battery 1A is model LCR-29 rated at 2030 ampere-hours at an 8-hour discharge rate to 1.75 V per cell; station batteries 1B and 2B are model LCY-35 rated at 2147 ampere-hours at an 8-hour discharge rate to 1.75 V per cell; and station battery 2A is model LCR-25 rated at 1800 ampere-hours at an 8-hour discharge rate to 1.75 V per cell.

The NRC staff reviewed the licensee's dc sizing calculations (SENH-13-001, "Station Service Battery 1A SBO Extended Coping Time Study," Version 1.0, SENH-13-002, "Station Service Battery 1B SBO Extended Coping Time Study," Version 1.0, SENH-13-003, "Station Service Battery 2A SBO Extended Coping Time Study," Version 1.0, SENH-13-004, "Station Service Battery 2B SBO Extended Coping Time Study," Version 1.0, MC-H-14-0021, "Station Service Battery 1A (1R42-S001A) Sizing and Voltage Profile," Version 1.0, MC-H-14-0022, "Station Service Battery 1 B (1R42-S001B) Sizing and Voltage Profile," Version 1.0, MC-H-13-0034, "Station Service Battery 2A (2R42-S001A) Sizing and Voltage Profile," Version 2.0, and MC-H-13-0035, "Station Service Battery 2B (2R42-S001B) Sizing and dc MOV Voltage Analysis," Version 2.0), which verified the capability of the dc system to supply power to the required loads during the first phase of the Hatch FLEX mitigation strategy. The licensee's evaluation identified the required loads and their associated ratings (ampere (A) and minimum required voltage) and the non-essential loads that would be shed within 75 minutes to ensure battery operation until power is restored to the battery chargers (approximately 10 hours). Based on its review, the NRC staff confirmed that Battery 1A could be extended up to 15.02 hours, Battery 1B could be extended up to 16.68 hours, Battery 2A could be extended up to 14.42 hours, and Battery 2B could be extended up to 15.97 hours by shedding non-essential loads.

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

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

The licensee's Phase 2 strategy includes repowering 600 Vac buses C (Division I) and D (Division II) on both units approximately 10 hours after initiation of an ELAP event. The licensee's strategy relies on a portable 545 kilowatt (kW) 600 Vac FLEX DG for each unit. The licensee has a total of three portable 600 Vac FLEX DGs. The 600 Vac FLEX DGs would provide power to two 125/250 Vdc battery chargers per unit, RCIC controls, control room coolers, RHR MOVs, and other selected loads.

The NRC staff reviewed the licensee's calculation A-47402, "FLEX Portable System 600V FLEX Diesel Generator Sizing Calculation," Version 1.0, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the EDGs. Based on the NRC staff's review, the required loads for the Phase 2 600 Vac 545 kW FLEX DG total approximately 387 kW running for both units' Division I loads and 383 kW running for both units' Division II loads. To reduce voltage and frequency dips, the licensee's procedures provide directions to sequence loads when loading the FLEX DGs. Therefore, one 600 Vac 545 kW FLEX DG is adequate to support the electrical loads required for the licensee's Phase 2 strategy.

If one of the "N" FLEX 600 Vac DGs becomes unavailable or is otherwise out of service for maintenance, the other ("N+1") FLEX 600 Vac DG would be deployed to continue to support the required loads. The "N+1" FLEX 600 Vac DG is identical to the "N" FLEX DGs, thus ensuring electrical compatibility and sufficient electrical capacity in an instance where substitution is required. Since the "N+1" FLEX 600 Vac DG is identical and interchangeable with the "N" FLEX 600 Vac DGs, the NRC staff concludes that the licensee has met the provisions of NEI 12-06, for spare equipment capability regarding the Phase 2 FLEX generators.

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 an NSRC includes four 1-megawatt (MW) 4160 Vac combustion turbine generators (CTGs), two 1100 kW 480 Vac CTGs, two 480 Vac to 600 Vac step-up transformers, and distribution panels (including cables and connectors). Since the licensee plans to continue its Phase 2 coping strategy, two 480 Vac CTGs and the 480 Vac to 600 Vac step-up transformers could be used to replace a Phase 2 600 Vac FLEX DGs, if necessary. Given that the NSRC-supplied 480 Vac CTGs are of greater capacity than the onsite Phase 2 FLEX DGs (1100 kW versus 545 kW), the NRC staff concludes that the NSRC-supplied 480 Vac CTGs have adequate capacity to supply the required Phase 3 loads, if necessary.

Additionally, each portable 4160 Vac CTG is capable of supplying approximately 1 MW, but two CTGs could be operated in parallel to provide a total of approximately 2 MW per unit. While the licensee does not anticipate needing to use the 4160 Vac CTGS supplied by an NSRC, they could power the opposite train that is powered during Phase 2 as defense-in-depth (i.e., either 4160 Vac switchgear E or G), if necessary. Licensee document number A-47390, "FLEX Engineering Judgement Review – FLEX Options Beyond 72 Hours," Version 1, included the licensee's basic strategy for using the NSRC-supplied 4160 Vac CTGs including a total loading estimate. The licensee's calculation determined that the maximum running loads for Unit 1 would be approximately 1363.18 kW while the maximum running loads for Unit 2 would be 1273.19 kW. The licensee's calculation assumed that all the loads would be sequenced on the CTG with the largest motor starting first followed by the remaining loads, thereby preventing overloading the CTGs. Based on its review of this calculation, the NRC staff concludes that the NSRC supplied 4160 Vac CTGs have adequate capacity to supply the Phase 3 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 RCS inventory during an ELAP event consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-1 and Appendix C summarize an 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 capable of exceeding the boil-off rate for the design basis heat load; 2) makeup via connection to spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load; 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.

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

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

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

3.3.1 Phase 1

In its FIP, the licensee stated that operators will monitor SFP level using the SFP instrumentation installed per Order EA-12-051. The operators are directed by FSGs to establish ventilation in the reactor building by opening the reactor refueling roof vents and the doors on the SFP refueling elevation and the 130 ft. elevation. The SFP water inventory will heat up from 110°F to 212°F during the first 12 hours for Unit 1 and Unit 2, respectively.

3.3.2 Phase 2

In its FIP, the licensee stated that Phase 2 will utilize the Altamaha River as the makeup source via the intake structure. This will occur around 10 hours after initiation of the ELAP event. The FLEX pump draws suction from the Altamaha River and the FLEX hoses from the discharge of the FLEX pump are connected to RHRSW piping in the intake structure valve pit. The RHRSW piping runs from the intake structure to the reactor building, in which the makeup water can be distributed for RPV, CST, and SFP makeup through FLEX connections inside the reactor building. The SFP makeup connections from the RHRSW piping are described below in Section 3.7.3.1 of this evaluation.

3.3.3 Phase 3

In its FIP, the licensee stated that the SFP makeup strategy for Phase 3 will utilize Phase 2 connections and NSRCs equipment as backup. Operators are provided direction for using NSRC equipment for indefinite SFP makeup through procedure TSG 31 EO-TSG-001-0, "Technical Support Guidelines," Revision 15.

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.

As described in the licensee's FIP, the licensee's Phase 1 SFP cooling strategy does not require any actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of the SFP. The operators are directed by FSGs to manually open the reactor refueling roof vents. An alternate ventilation path can be established by opening the stairwell doors at the SFP refueling floor and the 130 ft. elevation in the reactor building.

The licensee's SFP cooling strategy involves the use of the FLEX pumps and associated hoses and fittings with suction from Altamaha River. 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 of this evaluation. The staff's evaluation of the robustness and availability of the Altamaha River is discussed in Section 3.10.3 of this evaluation.

3.3.4.1.2 Plant Instrumentation

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

3.3.4.2 Thermal-Hydraulic Analyses

In its FIP, the licensee analyzed three different SFP cooling heat load cases when determining SFP makeup rate for Units 1 and 2. The heat load cases are documented in calculation SMNH-98-019, "Fuel Pool Time to Boil," Revision 2. These three cases are a maximum heat load during an outage following a normal batch discharge (Case 1), a maximum heat load during an outage following a core shuffle (Case 2), and a maximum heat load during an outage following a full core offload (Case 3). The maximum heat load, boil-off time to top of the fuel and makeup rate can be found in the table below (Unit 2 is listed since the time to boil is achieved earlier than Unit 1).

	Heat Load	Time to boil	Makeup rate
Case 1	10.02 million Btu/hr	13.91 hrs	20.8 gpm
Case 2	14.81 million Btu/hr	9.81 hrs	30.4 gpm
Case 3	34.425 million Btu/hr	4.22 hrs	71.3 gpm

As stated in Section 3.2.3.5 of this safety evaluation, the licensee's FLEX pump is capable of supplying 75 gpm of SFP makeup and 250 gpm of SFP spray flow to each SFP, which is more than the worst case SFP makeup requirements.

NEI 12-06, Section 3.2.1.6, states that one of the initial SFP conditions is that the SFP heat load assumes the maximum design-basis heat load for the site. Based on the NRC staff's review of the SFP time to boil calculations, the NRC staff concludes that Hatch has considered the maximum design-basis SFP heat load as part of the flow rate requirements for the FLEX pump.

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 Phases 2 and 3. In the FIP, Section 2.4.10.1 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX pump. The FIP states that the FLEX pump can provide nominally 3000 gpm at 150 psig discharge pressure, which includes 75 gpm for SFP makeup using the FLEX hoses. The staff noted 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 pump can provide SFP spray flow rate of 250 gpm for each unit, which exceeds the maximum SFP makeup requirements. The staff reviewed the SFP analysis described above and concludes that it is consistent with NEI 12-06, Section 11.2 and the FLEX equipment is capable of supporting the SFP cooling strategy.

3.3.4.4 Electrical Analyses

The licensee's mitigating strategies for the SFP do not rely on electrical power except for power to the SFP level instrumentation. The licensee's Phase 1 electrical SFP cooling strategy is to monitor SFP level using installed instrumentation (the capability of this instrumentation is described in other areas of this safety evaluation). The Hatch SFP level instrumentation has a backup battery with sufficient capacity to ensure a minimum of 24 hours of operation. Prior to the battery fully depleting, the licensee would restore power to the instrumentation using a 600 Vac FLEX DG.

The licensee's Phase 2 and 3 electrical SFP cooling strategy is to continue monitoring SFP level using installed instrumentation. During these Phases, the licensee would utilize the 600 Vac FLEX DGs and NSRC supplied CTGs to provide power to ensure indefinite SFP level monitoring capability. Based on its review of the licensee's FLEX DG sizing calculation (A-47402), the NRC staff concludes that the 600 Vac 545 kW FLEX DGs and NSRC supplied CTGs are adequately sized to ensure continued SFP cooling.

Based on its review, the NRC staff concludes that the licensee's electrical strategy appears acceptable to restore or maintain SFP cooling indefinitely during an ELAP event.

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. One such approach is for a licensee to perform an analysis demonstrating that containment pressure control is not challenged. Hatch has a General Electric BWR with a Mark I containment for both Unit 1 and Unit 2.

The licensee performed a containment evaluation, SMNH-12-032, "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 the suppression pool via the HCVS to maintain containment pressure and temperature within acceptable values and concluded that the Unit 1 containment pressure remains well below the respective UFSAR Section 5.2, Table 5.2-7 drywell and suppression chamber design limits of 56 psig for more than 72 hours. The analysis predicts the drywell temperature will exceed the design temperature limit of 281°F by 3°F. The licensee indicated that they will use drywell sprays via FLEX pumps to maintain Unit 1 drywell temperature below the design limit temperature. The Unit 2 containment pressure and temperature remain well below the respective UFSAR Section 6.2, Table 6.2-1 drywell and suppression chamber design limits of 56 psig and 340°F, respectively, for more than 72 hours. From its review of the evaluation, the NRC staff noted that the required actions to maintain containment integrity and required instrumentation functions have been developed, and are summarized below.

3.4.1 Phase 1

Hatch's Phase 1 strategy assumes containment isolation at the onset of the event. Procedures NMP-OS-019-266, "Hatch Unit 1 SIG-6, Containment Integrity," and NMP-OS-019-286, "Hatch Unit 2 SIG-6, Containment Integrity," provide guidance for assuring containment isolation. Phase 1 containment limitations are driven by the RCIC pump suction temperature limit of 230°F and the unsafe regions of the heat capacity temperature limit (HCTL) or SRV tail pipe limit. After the CST is depleted, or if the CST is unavailable, the RCIC pump is aligned to take suction from the suppression pool. Calculation SMNH-12-032 forecasts the suppression pool will reach 230°F after approximately 7 to 8 hours. At approximately 5 hours, the torus will be vented through the HCVS per guidance 31EO-FSG-003-1, "FSG - ELAP (Extended Loss Of AC Power)," along with guidance NMP-OS-019-266 for Unit 1 (31EO-FSG-003-2, "FSG - ELAP (Extended Loss Of AC Power)," and guidance NMP-OS-019-286 for Unit 2 to prevent initially entering the unsafe regions of the HCTL or SRV tail pipe limit. The rupture disk in the HCVS flow path is breached by pressurizing the HCVS piping upstream of the rupture disk to above its rated rupture point. This is accomplished using a pre-staged argon bottle from a local station. Venting the torus to maintain a suppression pool temperature of 230°F maintains a suppression chamber pressure less than 19 psig, well below the 56 psig design pressure.

3.4.2 Phase 2

The Phase 2 strategy maintains the implementation of Phase 1 with a FLEX generator providing power to instrumentation, valves, and other components of the HCVS.

3.4.3 Phase 3

There are no time sensitive Phase 3 actions that have been identified. The FIP indicates that as resources become available, actions can be taken to transition away from extended Phase 2 coping strategies. Instructions for connection and utilization of NSRC equipment for long-term coping or recovery will be provided by Technical Support Center (TSC) personnel, who will have assessed the condition of the plant and infrastructure, plant accessibility, and additional available off-site resources (both equipment and personnel) following the BDBEE. Licensee program document NMP-OS-019-002, "TSC Support for Beyond Design Basis Events," provides guidance for the TSC.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

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

3.4.4.1.1 Plant SSCs

Containment

The primary containment system houses the reactor pressure vessel, the reactor coolant recirculation system, and other branch connections of the RCS. The primary containment consists of the drywell, the suppression chamber that stores a large volume of water, a connecting vent system between the drywell and suppression chamber, isolation valves, a vacuum relief system, containment cooling systems, and other service equipment.

The drywell is a steel pressure vessel in the shape of a light bulb, and the suppression chamber is a torus-shaped steel pressure vessel located below and encircling the drywell. The primary containment system is designed to withstand the pressures resulting from a breach of the nuclear system process piping up to and including an instantaneous circumferential break of the reactor recirculation piping.

The reactor building encloses the reactor, reactor primary containment, auxiliary cooling systems, new-fuel storage vault, and spent-fuel storage pool. The reactor building provides secondary containment for the reactor and primary containment for auxiliary systems. The reactor building is a reinforced concrete structure, with structural steel framing. All exterior doorways in the reactor building are designed to resist the full effects of tornado winds, depressurization, and wind driven missiles.

The reactor building is a seismic Category I structure. The drywell and suppression chamber are classified as seismic Category I. Seismic Category I structures, components, and systems are designed to withstand the effects of the design-basis earthquake (DBE) and operating basis earthquake (OBE).

HCVS

The HCVS is a safety-related seismically qualified system which is protected from BDBEE hazards as defined in NEI 12-06. The HCVS vent path at Hatch consists of an 18-inch diameter wetwell and an 18-inch diameter drywell vent on each unit. The drywell vent exits the primary containment into the reactor building and proceeds down to the torus bay. Wetwell and drywell vent piping merges into a common header in the torus bay. Vent path for both wetwell and drywell exits the reactor building through an underground pipe. This pipe travels approximately 500 ft. from both units and combines in a mixing chamber at the base of the main stack. All effluents exit out the main stack. The HCVS flow path valves are air-operated valves (AOV) with air-to-open and spring-to-shut (i.e., the wetwell containment isolation valves and the HCVS inlet isolation valve). Opening the valves requires energizing an ac powered solenoid operated valve and providing motive air/gas. The HCVS is provided with a permanently installed power source and motive air/gas supply adequate for the first 24 hours of the event. Beyond the first 24 hours, FLEX generators will be used to maintain battery power to the HCVS components. The initial stored motive air/gas will allow for a minimum of eight valve operating cycles for the HCVS valve for the first 24 hours.

Containment (Drywell) Coolers

No time sensitive actions have been identified for maintaining containment integrity; however, containment coolers, when supplemented by portable equipment (i.e., pumps for cooling water and DG for powering the fans) delivered from off-site (NSRC), can be aligned to support maintaining containment integrity long-term. The drywell cooling system consists of coil units and recirculating fans. The function of the recirculating fans is to assist the fan coil units in mixing the drywell air, thus maintaining a uniformly even temperature throughout the drywell space. The function of the fan coil units is to remove the heat in the drywell by drawing the hot air in the space through cooling coils. In turn, the cooling coils are cooled by the primary containment (drywell) chilled water system. The primary containment (drywell) chilled water system consists of two chilled water recirculation pumps, two centrifugal chillers, a chemical addition tank, a chemical feed pump, an expansion tank, and a chilled water piping circulating chilled water to the drywell fan coil units. In the UFSAR, Section 3 indicates that containment coolers are seismically qualified components which are also protected from the effects of a design basis tornado and BDBEE 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 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 is available, the FIP states that key credited plant parameters, including these containment parameters, would be available using alternate methods. Licensee program document NMP-OS-019-254, "Hatch Unit C Sig-11 Critical Instrumentation," provides instructions to obtain readings for critical instrumentation during times when the instrument has no power (ac or dc).

3.4.4.2 Thermal-Hydraulic Analyses

As mentioned above, the licensee performed a containment evaluation, SMNH-12-032, "Containment Analysis of FLEX Strategies," to analyze the Hatch containment response during an ELAP. The analysis used MAAP, Version 4.0.5. The base case analyzed assumed RCIC and HPCI are started at one minute into the event. The HPCI is secured at four minutes. The RCIC provides injection flow to the RPV with suction from the CST. The RCIC suction is swapped from the CST to the suppression pool when the CST inventory is depleted. The analysis assumes that the torus is vented through the HCVS when the suppression pool exceeds 230°F or that the SRV tail pipe level limit is reached. The MAAP calculation predicts this to occur at approximately 7.5 hours. Procedurally, during an event, the HCVS will be conservatively opened at approximately 5 hours when the SRV tail pipe suppression pool level limit is exceeded. The RCIC suction is swapped back from the suppression pool to the refilled CST when the suppression pool reaches a low level limit (suppression pool level is reduced by evaporation of inventory via the opened HCVS). Recirculation pump seal leakage is modeled as 18 gpm per pump at full reactor pressure (36 gpm total pump seal leakage). The analysis modeled 72 hours. The MAAP analysis predicts the maximum suppression pool water temperature to spike a little above 230°F and stabilize at 226°F. The maximum suppression pool pressure is 16.4 psig. The maximum drywell pressure is 16.3 psig. This is well below the Unit 1 and Unit 2 drywell and suppression chamber design limits of 56 psig.

The model predicts the maximum drywell temperature of 284°F. This is below the Unit 2 drywell and suppression chamber design limit of 340°F. However, the predicted maximum drywell temperature exceeds the Unit 1 drywell design limit of 281°F by 3°F. The maximum temperature occurs at approximately 72 hours. As part of the audit process, the NRC staff requested additional information regarding operating at or above the Unit 1 design temperature. The licensee indicated that injection of water from the drywell spray via the FLEX pumps will be used to control the drywell temperature in Unit 1.

3.4.4.3 FLEX Pumps and Water Supplies

For Unit 1, FLEX pumps and water supplies may be used to maintain the drywell temperature below the design limit of 281°F. FLEX pumps and water supplies are not directly credited for maintaining Unit 2 containment functions. To support RCIC operation, the HCVS will be used to maintain the suppression pool temperatures below 230°F by venting steam from the suppression pool to the main stack. The suppression pool pressure is maintained below 19 psig. Venting the suppression pool also reduces the water level in the suppression pool. RCIC will be used for reactor core cooling until the suppression pool water level reaches the technical specification low level limit of 146 inches. At that time, the RCIC suction source is swapped back to the refilled CST from the suppression pool.

3.4.4.4 Electrical Analyses

The licensee performed a containment evaluation based on the boundary conditions described in Section 2 of NEI 12-06. Based on the results of its evaluation, the licensee developed required actions to ensure maintenance of containment integrity and required instrumentation continues to function. With an ELAP initiated, while Hatch is in Modes 1-4, containment cooling would be lost for an extended period of time. Therefore, containment temperature and pressure will slowly increase.

The licensee's Phase 1 coping strategy for containment includes monitoring containment temperature and pressure using installed equipment. The licensee also plans to use the HCVS throughout the event (starting approximately 5 hours after ELAP initiation). The power source for the HCVS valves is provided by two critical instrument cabinets fed from inverters which are powered from the 250 Vdc switchgears A and B, which in turn are connected to station service batteries A and B. The licensee's strategy to repower instrumentation using the Class 1E station batteries for Phase 1 is described in Section 3.2.3.6 of this safety evaluation and is adequate to ensure continued containment monitoring and HCVS operation.

The licensee's Phase 2 coping strategy is to continue the Phase 1 coping strategy. The licensee's strategy is to repower instrumentation and HCVS components using a 600 Vac FLEX DG. Use of the FLEX DG is described in Section 3.2.3.6 of this safety evaluation. The FLEX DG is adequately sized to ensure continued containment monitoring and HCVS operation.

The licensee's primary Phase 3 coping strategy is to continue using its Phase 2 coping strategy with offsite equipment used as backup, if necessary. The two 480 Vac CTGs and 480 Vac to 600 Vac step-up transformers could be used to replace the Phase 2 600 Vac FLEX DGs. Based on the licensee's analysis (SMNH-12-032, "Containment Analysis of FLEX Strategies (MAAP4 Calculation)," Version 3.0), drywell temperature should remain below the containment design-basis temperatures beyond 72 hours via actions of venting, injection of cool water from the river, and drywell sprays via FLEX pumps (see Section 3.9.1.1 of this safety evaluation for further details on equipment operation with loss of cooling). Hatch will receive offsite resources and equipment from an NSRC allowing implementation of long-term strategies to control containment pressure and temperature. Operators will continue to monitor containment parameters to inform the emergency director/TSC staff when additional actions may be required to reduce containment temperature and pressure. If RCIC remains in operation to provide core cooling, the drywell temperature will continue to increase. Therefore, it will become necessary to utilize a portable pump to provide containment cooling with torus or drywell spray. Whether a portable pump is in service for CST makeup or RPV injection, the core cooling strategy can be altered to provide containment cooling through valve manipulation. Licensee document No. A-47390 contains general guidance for the use of NSRC equipment to establish containment cooling.

3.4.5 Conclusions

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

3.5 Characterization of External Hazards

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

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

The licensee reviewed the plant site against NEI 12-06 and determined that FLEX equipment should be protected from the following hazards: seismic; severe storms with high winds; snow, ice and extreme cold; and extreme high temperatures. Regarding external flooding, Hatch is built above the design basis flood and is considered a "dry" site by the NEI 12-06 guidance.

References to external hazards within the licensee's mitigating strategies and this safety evaluation 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) (ADAMS Accession No. ML12053A340) (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 (80, FR 70610). 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" (ADAMS Accession No. ML14309A256). The Commission provided guidance in an SRM to COMSECY-14-0037 (ADAMS Accession No. ML15089A236). The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 (ADAMS Accession No. ML15174A257), the NRC staff informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the NRC safety evaluations 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 (ADAMS Accession No. ML16005A625). The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 (ADAMS Accession No. ML15357A163). The licensee's MSAs will evaluate the mitigating strategies described in this safety evaluation 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 safety evaluation 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 OBE and the DBE. As described in UFSAR Section 2.5, the OBE and DBE are .08g and .15g, respectively. Note that the current NRC terminology for the DBE is the safe shutdown earthquake (SSE). 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 Hatch is built above the design-basis flood level. Specifically, as described in UFSAR Chapter 2, the probable maximum flood elevation is 105 ft. MSL without wave runup and up to 108.3 ft. with wave runup. However, grade level at Hatch is 129 ft. MSL, and the floor elevation in the intake structure is 110 ft. MSL. Therefore, in accordance with NEI 12-06, Hatch is considered a "dry" site.

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 out this external hazard.

3.5.3 High Winds

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

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

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

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 31° 56' 2" North latitude and 82° 20' 39" West longitude. In NEI 12-06 Figure 7-2, Recommended Tornado Design Wind Speeds for the 1E-6/year Probability Level indicates that the site is in a region where the tornado design wind speed exceeds 130 mph. Therefore, the plant screens in for an assessment for high winds and tornados, including missiles produced by these events. Regarding hurricanes, Hatch is located where wind speed exceeds 130 mph; therefore, the site screens in for hurricanes.

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, and as stated above regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 31° 56' 2" North latitude and 82° 20' 39" West longitude. In addition, the site is located within the region characterized by EPRI as ice severity level 5 (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.

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

3.5.5 Extreme Heat

In the section of its FIP regarding the determination of applicable extreme external hazards, the licensee stated that, as per NEI 12-06 Section 9.2, all sites are required to consider the impact of extreme high temperatures. Summers at the site may bring periods of extremely hot weather over 100°F. The plant site screens in for an assessment for extreme high temperature hazard.

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

3.5.6 Conclusions

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

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

In its FIP, the licensee stated that Hatch has a single concrete structure (i.e., FSB) to store the FLEX equipment. In addition, the structure has a mezzanine for extra storage space. The storage building is located outside of the protected area onsite but within the owner controlled area. Below are additional details on how FLEX equipment is protected from each of the applicable external hazards.

3.6.1.1 Seismic

The Hatch storage building was constructed to meet the plant's DBE or SSE. In addition, large equipment stored in the building will be secured using tie-downs integrated into the floor slab to prevent seismic interaction.

3.6.1.2 Flooding

As mentioned above, Hatch is considered a "dry" site. In its FIP, the licensee stated that the FSB is located above the upper-bound flood stage elevation. In addition, the storage building was constructed to prevent water intrusion.

3.6.1.3 High Winds

As stated above, the Hatch storage building is a concrete structure, which is protected from tornado-missile events. The design of the building includes two personnel entry/exit doors and two equipment doors. Each of the doors are designed to the design-basis tornado wind pressure loads and tornado-missile loads.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee stated that the Hatch storage building includes a heating and ventilation system. The heating and ventilation system will ensure that the normal storage temperature conditions are suitable for long-term equipment reliability. With the heating and ventilation system, temperatures internal to the building will be maintained between 50°F and 100°F. Regarding ice, procedure 34AB-Y22-002-0 provides direction for managing potential winter weather events.

3.6.1.5 Conclusions

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

3.6.2 Availability of FLEX Equipment

Section 3.2.2.16 of NEI 12-06 states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare (i.e., an N+1 capability, where "N" is the number of components needed to support the safety function for all units at the 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, FLEX DGs, and support equipment identified in the FIP and during the audit review, the NRC staff concludes 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 guidance in Section 3.2.2.16 of NEI 12-06.

3.7 Planned Deployment of FLEX Equipment

The licensee stated in its FIP that multiple paths are available from the FSB to the deployment location to minimize the potential challenge from debris sources. The haul paths are shown in Figure 4 in the FIP. These haul paths have been reviewed for potential soil liquefaction and the licensee determined that the likelihood of liquefaction is low.

3.7.1 Means of Deployment

The deployment of onsite FLEX equipment to implement coping strategies beyond the initial plant capabilities (Phase 1) requires that pathways between the FSB and various deployment locations be clear of debris resulting from the applicable external hazard. In its FIP, the licensee stated that the stored FLEX equipment includes a heavy duty pickup truck and a small semi-tractor, which will ensure reliable towing/transport of the FLEX equipment from the storage location to the deployment areas.

Under normal circumstances, the licensee may need to open doors and gates that rely on electric power for opening and/or locking mechanisms. During the onsite audit, the licensee stated that access to the FLEX equipment and transport to the deployment locations do not require ac power. The licensee has contingencies for access upon loss of all ac/dc power as part of the security plan. Access to the owner-controlled area, the plant protected area, and areas within the plant structures will be controlled under this access contingency.

As stated above, the licensee has identified multiple paths from the FSB to the deployment locations to minimize the potential challenge from debris sources. After the onset of an ELAP, the licensee will complete an initial assessment of damage caused by the external hazard to allow the selection of which set of FLEX equipment to utilize and the most readily available transport path. However, high winds can cause debris from distant sources to interfere with planned haul paths. Therefore, tow vehicles and debris removal equipment is stored in the storage building, which protects the equipment from severe storm and high wind hazards such that the equipment remains functional and deployable to clear obstructions from the pathway between the storage location and its deployment location(s) in a timely manner. In addition, the licensee completed a walk down and determined that the haul paths can support a minimum of two lanes of vehicular traffic, which will decrease the likelihood of a path being completely blocked from debris, as well as reduce the time to clear debris to deploy equipment.

Phase 3 of the FLEX strategies involves the receipt of equipment from offsite sources including the NSRC and various commodities such as fuel and supplies. Transportation of this equipment can be through airlift or via ground transportation. Debris removal for the pathway between the site and the NSRC receiving locations for Hatch and from the various plant access routes may be required. The same debris removal equipment used for on-site pathways can be used to support debris removal to facilitate road access to the site.

3.7.2 Deployment Strategies

In its FIP, the licensee stated that the haul paths were evaluated for potential soil liquefaction. The licensee determined that the overall liquefaction potential at the FSB and across the designated travel paths is low for the postulated seismic ground motions. In addition, the risk of surface displacement due to faulting or lateral spreading is also deemed low. The NRC staff walked down and reviewed the licensee's travel paths during the onsite audit to verify the licensee's conclusions and the NRC staff believes that liquefaction should not inhibit the necessary equipment deployment after a seismic event.

For the RCS cooling and SFP makeup strategies, the licensee will deploy a portable diesel-driven FLEX pump to supply both units for CST makeup (for RPV injection via RCIC) or direct RPV injection, SFP portable spray monitors, main control room cooling, and RCIC room cooling. The FLEX pump will be deployed at the intake structure and two submersible pumps will take suction from the Altamaha River. The submersible pumps will feed the FLEX pump, which will discharge to the RHRSW piping using flexible hoses.

For the electrical strategy, the licensee will deploy two FLEX 600 Vac DGs into the protected area. One 600 Vac DG will be staged at the control building west wall and will power bus C on Units 1 and 2. The other 600 Vac DG will be staged south of the Unit 2 turbine building and will power bus D on Units 1 and 2. The alternate strategy consists of powering the 600 V battery chargers directly from the FLEX DGs, via temporary cables, to FLEX permanent transfer switch/receptacle units near the chargers.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling

In its FIP, the licensee stated that the primary connection for the FLEX pump is at the intake structure. The suction of the FLEX pump is supplied by submersible pumps to draw makeup water from Altamaha River. Inside the intake structure valve pit, five lines of piping are connected into the 18-inch RHRSW piping. FLEX hoses will be run from the discharge of the

FLEX pump into the five lines for distribution to the RHRSW system, which can support RPV injection or CST makeup. The connection points at the intake structure are protected from all applicable external hazards. For RPV makeup, the RHRSW piping in the intake structure runs to the reactor building, where the RHRSW piping is cross-tied with the RHR system to allow Phase 2 RPV injection using either division of RHR. The alternate RPV pathway allows for the operators to route a hose from the FLEX pump directly into the RHRSW piping in the reactor building. The entry point for the alternate core cooling makeup will be located at two existing reactor building penetrations, one for each unit. For CST makeup, the RHRSW piping in the reactor building allows for the existing CST core spray system suction line to receive makeup water for each unit using the FLEX pump. The FLEX connection piping has an isolation valve and check valve to prevent accidental draining of the CST.

SFP Cooling

In its FIP, the licensee stated that the primary connection for SFP makeup is the RHRSW piping in the intake structure via the FLEX pump. The RHRSW piping leads into the reactor building, where a new FLEX connection will be located on the SFP refueling floor. The new FLEX connection can allow either a hose connection to provide makeup directly to the SFPs or a connection to the spray monitor nozzles. An additional connection is a crosstie from the RHRSW to the PSW system piping in the reactor building, which provides an emergency fill connection to the SFP makeup cooling piping. Operators are directed to isolate valves to allow the SFP cooling piping to be accessible from the injection into the PSW system piping. The alternate connections for SFP makeup are located at two penetrations in the reactor building, one for each unit. Storz adaptors and braided hoses will be used to connect to the new FLEX header located on the SFP refueling floor. The SFP makeup connections located in the reactor building and intake structure are protected from all applicable external hazards as defined in NEI 12-06.

3.7.3.2 Electrical Connection Points

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

During Phase 2, the licensee's strategy is to supply power to equipment required to maintain or restore core cooling, containment, and SFP cooling using a combination of permanently installed and portable components.

The licensee's primary electrical strategy is to connect the 600 Vac FLEX DG to the 600 Vac FLEX connection boxes that are protected against the site's applicable external hazards. The 600 Vac FLEX connection boxes are connected to safety-related/seismically qualified breakers via conduit and cable that would power bus C (Division I) and bus D (Division II) on each unit. Powering these buses provide the ability to power battery chargers A/B or C and D/E or F in order to charge the station batteries and supply dc loads.

The 600 Vac FLEX connection boxes are located in the west cableway underneath the west side of the Unit 1 control building (Division I), and underneath the south wall of the Unit 2 turbine building (Division II). The connection boxes provide color-coded connections for a 600 Vac FLEX DG to be plugged into the system.

The staging area for one 600 Vac FLEX DG would be outside the service building near the Unit 1 control building's west wall. The second 600 Vac FLEX DG would be staged near the south side of the Unit 2 turbine building. Each FLEX DG will power one division on both units. Having each 600 Vac FLEX DG powering a division on both units provides the capability of powering at least one division in each unit should one of the two FLEX DG connection points become unavailable (i.e., deploying two 600 Vac FLEX DGs and utilizing multiple diverse connection points should ensure that two divisions per unit will be powered). Therefore, primary and alternate connections and staging areas are included in the licensee's general strategy.

Alternatively, the 600 Vac FLEX DGs could be connected to the Division I and II battery chargers through four of the six seismically mounted receptacles for each unit. The licensee installed safety-related, seismically qualified manual transfer switches for each battery charger to provide the connection. Temporary cables would be brought from a 600 Vac FLEX DG through either the main door on the north side of the service building, roll up doors on the Unit 1 and Unit 2 turbine buildings, or the air lock in Unit 1's reactor building and through the control building, and would be connected to two of the three battery chargers in each division.

Program Document NMP-OS-019-262, "Hatch Unit 1 SIG-2, 600V Alternate Power," Version 5.0, and NMP-OS-019-282, "Hatch Unit 2 SIG-2, 600V Alternate Power," Version 5.0, provide direction for deploying and staging the 600 Vac FLEX DG. These procedures also provide guidance for connecting, making electrical alignments, and operating the 600 Vac FLEX DGs.

The licensee performed acceptance testing for the installed FLEX connectors that verified proper termination at each connector and verified that the phase rotation matched the existing plant configuration.

For Phase 3, the licensee will receive four 1 MW 4160 Vac CTGs and two 1100 kW 480 Vac CTGs, and a 480 Vac to 600 Vac step-up transformer from an NSRC. Since the licensee plans to continue its Phase 2 coping strategy, the 480 Vac CTGs and 480 Vac to 600 Vac step-up transformers could be used to replace the Phase 2 600 Vac FLEX DGs. The licensee would stage the CTGs near the Phase 2 FLEX DG it would be replacing and utilize NMP-OS-019-262 and NMP-OS-019-282, for connecting, and making the appropriate electrical alignments. The licensee would utilize these procedures and NMP-OS-019-002, "TSC Support for Beyond Design Basis Events," Version 2.0, to verify proper phase rotation for the Phase 3 CTGs.

If necessary, the 4160 Vac CTGs could be connected to the train opposite of the one powered during Phase 2 using an alternate supply breaker. For example, if the Phase 2 600 Vac FLEX DG is connected to load centers 1D and 2D, the Phase 3 4160 Vac CTGs could be connected to 4160 Vac Buses 1G and 2G. Program documents NMP-OS-019-261, "Hatch Unit 1 SIG-1, 4160V Alternate Power," Version 1.0, and NMP-OS-019-281, "Hatch Unit 2 SIG-1, 4160V Alternate Power," Version 1.0, provide guidance on deploying and connecting the 4160 Vac CTGs. These procedures and NMP-OS-019-002 includes guidance for verifying proper phase rotation prior to energizing plant equipment. The primary staging area would be outside of the EDG building while the alternate staging area would be the east side outside of Unit 2 reactor building.

3.7.4 Accessibility and Lighting

During the onsite audit, the licensee stated that the potential impairments to required access are: 1) doors and gates, and 2) site debris blocking personnel or equipment access. The coping strategy to maintain site accessibility through doors and gates is applicable to all phases of the FLEX coping strategies, and is immediately required as part of the immediate activities required during Phase 1. Doors and gates serve a variety of barrier functions on the site. One primary function is security and is discussed below. However, other barrier functions include fire, flood, radiation, ventilation, tornado, and high energy line break. As barriers, these doors and gates are typically administratively controlled to maintain their function as barriers during normal operations.

The licensee noted that following an BDBEE and subsequent ELAP event, FLEX coping strategies require the routing of hoses and cables to be run through various barriers in order to connect beyond-design-basis (BDB) equipment to station fluid and electric systems or require the ability to provide ventilation. For this reason, certain barriers (gates and doors) will be opened and remain open. This deviation of normal administrative controls is acknowledged and is acceptable during the implementation of FLEX coping strategies. The ability to open doors for ingress and egress, ventilation, or temporary cables/hoses routing is necessary to implement the FLEX coping strategies.

In its FIP, the licensee described that the majority of areas for ingress/egress and deployment of FLEX strategies have safe shutdown battery operated lighting, which will provide 8 hours of operation after an ELAP. However, flashlights are the primary credited means of lighting to accomplish FLEX actions. In addition, while not credited in the FLEX strategies, large FLEX equipment include portable lighting to support actions in the yard outside of the protected area. Regarding main control room lighting, the licensee stated that an analysis was performed and determined that after load shed actions are completed the control room lighting should be reliable for a minimum of 14 hours or until the 600 Vac DG is available.

3.7.5 Access to Protected and Vital Areas

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

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee stated that the five underground diesel fuel oil storage tanks (DFOST) each have 40,000 gallons of diesel fuel oil, which will supply all of the diesel-driven FLEX equipment over 72 hours. The DFOSTs are seismically qualified and missile protected. The DFOST transfer pumps are repowered as part of the Phase 2 strategy from the 600 Vac FLEX DG to supply diesel fuel oil to the FLEX fuel tanker deployed from the FSB. Operators are directed by FSGs to connect hoses from the DFOST fill valve to the FLEX fuel tanker. An alternative connection point for refilling the FLEX fuel tanker is located in the EDG building day tank room. The licensee has three FLEX tanks and trailers, stored in the FSB, that are used to transport the fuel from the DFOSTs to the FLEX equipment. The haul routes for supplying the fuel will be the same as the deployment routes of the FLEX equipment.

The quality of fuel oil in the DFOSTs is maintained in accordance with the site's technical specifications. Fuel oil in the fuel tanks of portable diesel-driven FLEX equipment will be maintained in the preventative maintenance program in accordance with the manufacturer's guidance and existing site maintenance practices. The licensee also stated during the audit that the ultra-low sulfate diesel (ULSD) fuel will be stored in the FSB for refueling of FLEX equipment that uses ULSD fuel only. The licensee stated that two of the three fuel trailers will be loaded with ULSD fuel to ensure sufficient fuel is stored in the FSB to service FLEX equipment requiring ULSD fuel only. The remaining trailer would be used for transferring the normal diesel fuel from the DFOSTs to all remaining FLEX equipment. The licensee stated during the audit that after 72 hours, existing agreements are in place with suppliers, and the licensee maintains a large quantity of fuel oil available throughout the region and at two other nuclear facilities such that the supplemental fuel oil supplies should be available. Therefore, the diesel fuel oil onsite should be maintained such that the diesel-driven equipment will be available during an ELAP.

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 Hatch SAFER Plan

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

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

By letter dated September 26, 2014 (ADAMS Accession No. ML14265A107), 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 Hatch, Alternate Staging Area D is not used. Staging Area C is the Tattnall County/Reidsville, Georgia airport. Staging Area B is a large hard-surfaced area approximately 2-3 acres in size located within the owner controlled area adjacent to the site warehouse. Staging Area A is the final deployment location at the site.

Use of helicopters to transport equipment from Staging Area C to Staging Area B is recognized as a potential need within the Hatch SAFER Plan.

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

The FIP notes that following a BDBEE and subsequent ELAP event at Hatch, ventilation providing cooling to occupied areas and areas containing FLEX strategy equipment will be lost. The heat up of these areas will be caused by heat released by components operating on backup power, or sensible heat from components which were operating prior to the event but lost power due to the ELAP. The resulting temperature increase can result in vital areas exhibiting excessive temperatures which can impact accessibility and equipment functionality. Hatch operators are trained on working in high temperature areas of the plant. In addition, current general site training includes a module on the recognition of dehydration along with methods to cope. Bottled water is stored onsite. Guidance derived from current site industrial safety procedures and passive cooling technologies already used by response personnel will be applied as deemed necessary to minimize adverse impacts of heat stress.

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 acceptable for personnel habitability or accessibility and within equipment limits. The key areas identified for all phases of execution of the FLEX strategy activities are the main control room, RCIC pump room, Class 1E battery and switchgear rooms, and containment.

Main Control Room

Licensee calculation SMNH-12-031, "Hatch Main Control Room Heat up Evaluation During an Extended Loss of all ac Power," Version 1.0, determined that mitigating actions are required to ensure that the main control room temperature remains below 110°F. Under ELAP conditions with no mitigating actions taken, the licensee's analysis determined that the main control room would surpass 110°F approximately 9 hours after initiation of an ELAP event.

The licensee's Phase 1 FLEX strategy includes blocking open specific doors at or before 3 hours following ELAP initiation. This establishes a ventilation path from the control building (and outside) 130 ft. elevation to the main control room. In addition, the turbine building tornado roof vent hatches can be opened to allow hot air from the turbine building to escape. Allowing hot air to escape from the turbine building will aid in slowing the main control room heatup. Program document NMP-OS-019-255, "Hatch Unit C MCR Ventilation," Version 2.0, provides operator guidance for establishing main control room ventilation.

If the outside temperature is above 96°F, then operators would not open certain doors until the control room temperature is in excess of the outside temperature. According to the licensee's FIP, external temperatures normally only exceeds 98°F for a limited time during the early afternoon hours. In addition, according to the licensee's analysis, there is on average a 23.8°F difference between the daily high and low temperatures.

For Phase 2, the licensee's primary strategy for maintaining the environment of the main control room is to power one division of the control room chillers and air handling units. This requires the 600 Vac switchgear to be energized with a 600 Vac FLEX DG and to provide a jumper from the RHRSW to the PSW piping that supplies cooling water to the control room air conditioning coolers. Cooling is established no later than 13 hours following initiation of an ELAP. The main control room air conditioners are common to both units, and cooling water would be supplied to each main control room from both units. The FLEX connection and piping coming off of the 18 inch RHRSW line provides the means to supply a total of 120 gpm each to two of the main control room air conditioning coolers from both units. Valve manipulations are required to properly route the cooling water to the air conditioning units. The alternate strategy for Phase 2 is to power the opposite division of the control room coolers and air-handling units. Program document NMP-OS-019-255 provides guidance for restoring main control room chillers and air handling units.

The licensee's Phase 2 strategy should ensure that temperatures remain below habitability and equipment limits.

Based on the licensee's analysis and the availability of procedures to maintain temperatures 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 concludes that the electrical equipment in the main control room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

RCIC Pump Rooms

During operation, there will be a considerable heat load within the RCIC pump rooms from the steam turbine and associated piping. SMNH-13-002, "RCIC Room Heat-Up During an Extended Loss of all ac Power (FLEX)," Version 1.0, contains the licensee's evaluation of the operation of RCIC without forced ventilation during an ELAP event. The design area temperature limit for RCIC is 295°F as listed in the Hatch UFSAR. The Hatch UFSAR also notes that for long-term operation (6 months), the safety-related components of the RCIC room are designed to operate with area temperatures of 148°F.

The licensee's evaluation showed that with no supplemental ventilation, the room would remain below 148°F during the initial 72 hours. Cooling of the RCIC pump rooms beyond 72 hours would be accomplished as necessary by repowering the RCIC pump room coolers and by providing cooling water via the RHRSW to PSW crosstie. In addition, Hatch will receive offsite resources and equipment from an NSRC following 72 hours after the onset of an ELAP event. Therefore, based on the temperature remaining below the design limits specified in the Hatch

UFSAR, the NRC staff concludes that the electrical equipment in the RCIC pump rooms will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Class 1E Battery and Switchgear Rooms

Licensee calculation SMNH-14-006, "Hatch Station Battery Room Heat up Evaluation During an Extended Loss of All ac Power," Version 1.0, determined that the maximum temperature for the station battery rooms (1 A, 1 B, 2A, and 28) during an ELAP will not exceed the acceptance criteria of 122°F for a period of 72 hours.

The licensee determined that during battery charging operations in Phases 2 and 3, in support of maintaining power to instrumentation and controls for core cooling, containment and SFP cooling functions, ventilation will be required in the main battery rooms for cooling the rooms and venting hydrogen released from the batteries during charging. The licensee's primary ventilation strategy is to repower the existing emergency exhaust fans that are connected to the emergency power bus. This will occur after a 600 Vac FLEX DG has been connected to power a 600 Vac bus. The licensee's alternate strategy is to open doors and set up portable fans within 72 hours.

While the licensee's ventilation strategy should maintain the battery rooms below the manufacturer's design limit (122°F), the elevated temperature will have an impact by increasing the charging current required to maintain the float charging voltage set by the battery charger. The elevated charging current will in turn increase cell water loss through an increase in gassing. Based on this, periodic water addition may be required or the float charging voltage reduced per the guidance contained in the C&D Technologies vendor manual.

In Phase 2, following the energization of some of the 600 Vac switchgear by the 600 Vac FLEX DGs, the rooms begin to heat up. In SMNH-13-005, "Hatch Switchgear Room on Control Building Elevation 130 Heatup Evaluation During an Extended Loss of all ac Power (ELAP)," Version 2.0, the licensee determined that opening doors and operating fans is adequate to maintain switchgear room temperatures below 122°F to support continued operation of the equipment. NUMARC 87-00 indicates that certain classes of electrical equipment (such as those in the dc equipment rooms at Hatch) will likely remain operable in thermal environments of 150°F to 300°F for up to 8 hours. The licensee's analysis demonstrated that the maximum expected temperature in any of the rooms where the inverters (including the new hardened vent inverters) are installed will not exceed 122°F (the temperature limit for the inverters). For the rest of the switchgear rooms, the licensee determined that the maximum temperature should remain below 125°F before select doors are opened at 3 hours and fans are deployed at 10 hours to establish ventilation to reduce temperature. The licensee's general expectation is to maintain temperatures below 120°F in all areas containing required electrical equipment unless their evaluations justify higher temperatures.

As stated above, Hatch will receive offsite resources and equipment from an NSRC that can be used to supplement cooling of plant equipment.

NMP-OS-019-270, "Hatch Unit 1 SIG-10, Ventilation," Version 1.0, and NMP-OS-019-290, "Hatch Unit 2 SIG-10, Ventilation," Version 1.0, provide guidance for establishing ventilation in the Class 1E Battery and Switchgear Rooms.

Based on the licensee's analysis and the availability of procedures to maintain temperatures below 122°F (the design limit of electrical equipment, including the batteries at Hatch), the NRC staff concludes that the electrical equipment in the Class 1E battery and switchgear rooms should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Containment

In accordance with NEI 12-06, the containment is assumed to be isolated following the event. As the suppression pool (or torus) heats up and begins to boil, the containment will begin to heat up and pressurize. Licensee calculation SMNH-12-032 determined that the containment design limits (56 psig for both units and 281°F (Unit 1)/340°F (Unit 2)) would not be exceeded. The calculation determined that the maximum drywell temperature would briefly reach 273°F at approximately 52 and 71 hours (during SRV opening) and the maximum drywell pressure would reach 16.3 psig at approximately 7 hours. The licensee's calculation determined that the Unit 1 and Unit 2 temperature limits would not be exceeded during an ELAP event. Operators would control drywell and suppression pool pressure by opening the wetwell vent when the suppression pool level reaches the SRV tail pipe level limit. The wetwell vent sizing allows the wetwell pressure to drop to nearly atmospheric. Actual wetwell pressure oscillates around 2.5 psig due to SRV cycling. Operators would control suppression pool temperature by venting which will ensure that temperature limits are not exceeded.

Beyond 72 hours, the licensee plans to maintain drywell temperature below the containment design basis by venting, injection of cool water from the river, and drywell sprays via FLEX pumps.

Hatch has 11 solenoid operated SRVs, but just 1 SRV is necessary to control the pressure gradient after the first few hours of the event. The Hatch SRV solenoids were tested under the following conditions: 355°F, 68 psig, 3 hours; 335°F, 50 psig, 3 hours; 265°F 44 psig, 18 hours; 215°F 11 psig, 101.8 days. Based on the test results and the redundancy provided by the 11 SRVs, the NRC staff concludes that pressure control function should remain available for an extended period of time. In addition, while it is not anticipated that the SRVs will malfunction, operators could use RPV depressurization, use of additional FLEX pump capacity for drywell spray, and use NSRC large capacity pumps to reduce the drywell temperatures, if necessary.

Since it is necessary to ensure the capability of the SRVs to perform the pressure relief function, the licensee would vent containment to reduce suppression pool inventory and containment pressure. The licensee has enhanced the HCVS to ensure required vent operations. At approximately 5 hours, operators would use the HCVS to provide long-term core cooling capability thru RCIC and to maintain containment parameters within limits.

Monitoring of containment (drywell) pressure and temperature is available via permanently installed plant instrumentation. The required instrumentation located within containment meet Regulatory Guide 1.97 category I design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display. These design requirements bound the expected environmental conditions during an ELAP; therefore required instrumentation should not be adversely impacted by a loss of ventilation as a result of an ELAP. Nonetheless, plant operators will continue to monitor containment parameters to inform the emergency director/TSC staff when additional actions may be required to reduce containment temperature and pressure. Licensee document No. A-47390 contains general guidance for the use of NSRC equipment to establish containment cooling, if necessary.

Based on temperatures remaining below the design limits of equipment for 72 hours and the availability of offsite resources to supplement cooling thereafter, the NRC staff concludes that the electrical equipment in the containment should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

3.9.1.2 Loss of Heating

In its FIP, the licensee stated that the normal daily minimum temperature ranges from 37°F at Macon in January to 71°F in July. An extreme minimum temperature of 3°F was recorded at Macon in January 1966. The NRC staff also checked the American Society of Heating, Refrigeration, and Air Conditioning Engineers (ASHRAE) weather database and noted that in Macon the ambient temperature exceeds 21°F 99 percent of the time. About 42 days per year have minimum temperatures below freezing. The licensee does not consider extreme cold to be a significant concern for the site. Heat tracing has been provided to the five 6 inch FLEX ball valves (on each unit) located at the intake structure.

The CST will be the initial source of water and the need for a backup supply is not anticipated prior to 19 hours into the BDBEE due to the capacity of the CST and suppression pool. When heat tracing is lost during cold weather events, it should not be necessary to thaw any of the FLEX connection in the relatively short length of time prior to connecting the FLEX pump. Heat tracing is not required to be maintained following a BDBEE.

The Hatch Class 1E station battery rooms are located substantially internal to the plant and would not be exposed to extreme low temperatures. At the onset of the event, the Class 1E battery rooms would be at their normal operating temperature and the temperature of the electrolyte in the cells would build up due to the heat generated by the batteries discharging and during recharging. Temperatures in the battery and switchgear rooms are not expected to be sensitive to extreme cold conditions due to their location, the concrete walls isolating the rooms from the outdoors, and lack of forced outdoor air ventilation during early phases of the ELAP event.

Based on the above, the NRC staff concludes that the Hatch Class 1E station batteries should perform their required functions as a result of loss of normal heating during an ELAP event.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3 is the potential buildup of hydrogen in the battery rooms as a result of loss of ventilation during an ELAP event. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. Licensee calculation SMNH-14-001, "FLEX Battery Room Hydrogen Generation," Version 1.0, concluded that the earliest time to accumulate a 2 percent hydrogen concentration in any one of the battery rooms is approximately 73 hours and the latest time is 99 hours. The licensee's primary ventilation strategy is to repower the existing emergency exhaust fans that are connected to the emergency power bus. This would occur after a 600 Vac FLEX DG has been connected to power a 600 Vac bus (which would occur well before 73 hours). The licensee's alternate strategy is to open doors and set up portable fans within 72 hours. Program document NMP-OS-019-270 and NMP-OS-019-290 direct operators to establish ventilation in the Class 1E battery rooms.

Based on its review of the licensee's battery room ventilation strategy, the NRC staff concludes that hydrogen accumulation in the Hatch vital battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP event.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

The licensee performed calculation SMNH-12-031, "Hatch Main Control Room Heat Up Evaluation During an Extended Loss of all ac Power," Version 1.0. The calculation used the Generation of Thermal-Hydraulic Information for Containment (GOTHIC) Version 6.0 thermal-hydraulic analysis computer program. The analysis predicted that under ELAP conditions with no mitigating actions taken, the control room would surpass 110°F (the assumed maximum temperature for efficient human performance as described in NUMARC 87-00) in approximately 9 hours. The Phase 1 FLEX strategy mitigating actions involves opening selected doors within the first 3 hours of an ELAP. Procedures 34AB-R22-003-1 and 34AB-R22-003-2, "Station Blackout," provide initial guidance for mitigating actions to establish natural ventilation paths.

According to the FIP, the Phase 2 primary strategy for maintaining the environment of the main control room is to power one division of the control room water chillers and air handling units when the 600 Vac switchgear is energized with the 600 Vac FLEX DG, and a jumper is provided from RHRSW to the PSW piping that supplies cooling water to the main control room air conditioning coolers no later than 13 hours following the BDBEE. The main control room air conditioning units are common to both units, and cooling water is supplied to each control room from both units.

Procedure NMP-OS-019-255, "Hatch Unit C MCR Ventilation," provides guidance to minimize heat-up of the main control room until normal power and cooling can be restored to the control room. This includes opening doors to establish natural draft ventilation and the Phase 2 strategy of restoring the main control room air conditioners to service following restoration of power to MCC's 1R24-S002, 1R24-S003 and/or 1R24-S029 via a FLEX portable 600 Vac generator to the U-1 600 Vac emergency buses 1C or 1D.

The NRC staff reviewed the calculation and procedures and determined that with the compensatory actions, personnel in the control room should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

3.9.2.2 Spent Fuel Pool Area

The FIP states that SFP bulk boiling will create adverse temperature, humidity, and condensation conditions in the SFP area. This will require a ventilation pathway to vent the humid atmosphere from the SFP area. The primary pathway will be established by manually opening the reactor refueling roof vents. An alternate ventilation path can be established by opening doors that allow steam to escape through the air lock doors. In order to establish flow of air through the SFP area, it is necessary to open stairwell doors at the refuel floor elevation and the 130 ft. elevation. Both of these strategies are provided in the technical support guidelines (TSG) (31EO-TSG-001-0, Technical Support Guidelines, Attachment 20). Establishing the vent path will occur prior to the time that the SFP commences boiling, no sooner than 12 hours into the event when in Modes 1-4, and 4.2 hours when in Mode 5. Guidance for establishing natural draft ventilation is provided in NMP-OS-019-270, "Hatch Unit 1 SIG-10, Ventilation," and NMP-OS-019-290, "Hatch Unit 2 SIG-10, Ventilation."

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

3.10.1 RPV Make-Up

The licensee described in its FIP that the RCIC pump will provide the initial cooling water to the RPV. The CSTs (one for each unit) are the suction source for the RCIC pump at the initiation of the ELAP event. The lower portion of the seismically qualified CST is protected from tornado missiles. Each CST is credited with 100,000 gallons of water to supply the RPV for 6.8 hours, before the suction source is swapped to the suppression pool (torus).

3.10.2 Suppression Pool Make-Up

The licensee described the torus in its FIP as a safety-related seismically qualified steel shaped pressure vessel located below and encircling the drywell. Each torus contains at least 665,000 gallons of water and will provide 12 hours of additional RPV makeup once the CST is depleted. The torus is used as the water source for RPV makeup as long as the RCIC pumps are available during the ELAP event.

3.10.3 Spent Fuel Pool Make-Up

The licensee described the SFP makeup water source as the FLEX pump taking suction from the Altamaha River and connecting to the RHRSW piping in the intake structure valve pit. The RHRSW piping leads into the reactor building to a new FLEX connection on the SFP refueling floor. The Altamaha River has an infinite amount of water inventory and is protected from all applicable external hazards as defined in NEI 12-06.

3.10.4 Containment Cooling

FSAR Section 9.4.6 indicates that the safety-related PSW is the water source for the containment coolers used to remove heat in the HPCI, RHR and containment spray pump rooms.

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, about 18 hours are available to implement makeup before boil-off results in the water level in the SFP dropping far enough to about 15 ft. above the fuel assemblies, and the licensee has stated that they have the ability to implement makeup to the SFP within that time.

When a plant is in a shutdown mode in which steam is not available to operate a steam-powered pump such as RCIC (which typically occurs when the RPV has been cooled below about 300°F), another strategy must be used for decay heat removal. The NRC-endorsed strategy is described in NEI 12-06. Section 3.2.3 provides guidance to licensees for reducing shutdown risk by incorporating FLEX equipment in the shutdown risk process and procedures. Considerations in the shutdown risk assessment process include maintaining necessary FLEX equipment readily available and potentially pre-deploying or pre-staging equipment to support maintaining or restoring key safety functions in the event of a loss of shutdown cooling. In the FIP, the licensee stated that it would follow this guidance. During the audit process, the NRC staff observed that the licensee had made progress in implementing this guidance.

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

3.12 Procedures and Training

3.12.1 Procedures

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

The licensee indicated that the FSGs at Hatch were developed to provide pre-planned strategies for accomplishing specific tasks associated with implementation of FLEX strategies. Strategy implementation guides (SIGs) were developed to have operator actions in the field included in a separate "operator friendly" procedure format. The FSGs and SIGs together are equivalent to the generic FSGs. The licensee stated that validation has been accomplished in accordance with industry developed guidance to ensure that the required tasks, manual actions and decisions for FLEX strategies are feasible and may be executed within specified time constraints.

3.12.2 Training

In its FIP, the licensee stated that initial training has been provided and periodic training will be provided to the target population responsible for implementing the beyond-design-basis emergency response strategies and implementing guidelines. In addition, personnel assigned to the direct execution of mitigation strategies for BDBEEs have received the necessary training to ensure familiarity with the associated tasks, instructions, and mitigating strategy time constraints. The training plan development was done in accordance with the Systematic Approach to Training (SAT).

3.12.3 Conclusions

Based on the description above, the NRC staff concludes 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 dated October 3, 2013 (ADAMS Accession No. ML13276A573), which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 (ADAMS Accession No. ML13276A224), 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. Preventative maintenance templates for the major FLEX equipment have also been issued.

In its FIP, the licensee stated that Hatch will maintain the onsite FLEX equipment with preventive maintenance and testing based on the generic EPRI industry program for maintenance and testing of FLEX equipment, as endorsed by the NRC staff on October 7, 2013.

In the absence of an EPRI FLEX template, existing maintenance templates were used to develop the specific maintenance and testing guidance. For all other equipment not covered by a maintenance template, manufacturer OEM or industry standards were used to determine the recommended maintenance and testing.

Based on the use of the endorsed program, which establishes and maintains a maintenance and testing program in accordance with NEI 12-06, Section 11.5, the NRC staff concludes that the licensee appears to have adequately addressed equipment maintenance and testing activities associated with FLEX equipment.

3.14 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 27, 2013 (ADAMS Accession No. ML13059A389), the licensee submitted its OIP for Hatch, in response to Order EA-12-051. By letter dated August 1, 2013 (ADAMS Accession No. ML13203A118), the NRC staff sent a Request for Additional Information (RAI) to the licensee. The licensee provided a response by letter dated August 29, 2013 (ADAMS Accession No. ML13242A293). By letter dated October 28, 2013 (ADAMS Accession No. ML13294A529), the NRC staff issued an ISE and RAI to the licensee.

By letters dated August 27, 2013 (ADAMS Accession No. ML13240A236), February 26, 2014 (ADAMS Accession No. ML14057A778), August 26, 2014 (ADAMS Accession No. ML14239A298), February 26, 2015 (ADAMS Accession No. ML15057A273), August 27, 2015 (ADAMS Accession No. ML15239B262), February 25, 2016 (ADAMS Accession No. ML16056A545), and August 8, 2016 (ADAMS Accession No. ML16221A387), 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 SFP level instrumentation which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated February 13, 2017 (ADAMS Accession No. ML17044A414), the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed a SFP level instrumentation system designed by Westinghouse. The NRC staff reviewed the vendor's SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report on August 18, 2014 (ADAMS Accession No. ML14211A346).

The staff performed an onsite audit to review the implementation of SFP level instrumentation related to Order EA-12-051. The scope of the audit included verification of the (a) site's seismic and environmental conditions enveloped by the equipment qualifications, (b) equipment installation met the requirements and vendor's recommendations, and (c) program features met the requirements. By letter dated January 13, 2016 (ADAMS Accession No. ML15349A801), the NRC issued an audit report on the licensee's progress. Refer to Section 2.2 above for the regulatory background for this section.

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

In its OIP, the licensee stated that Level 1 is the low level alarm setpoint which is at elevation 225 ft. 9 inches (in.) for Unit 1 and 226 ft. 2.5 in. for Unit 2. The licensee stated that this elevation is above the elevation where the pumps lose suction from a low level in the skimmer surge tank.

In its letter dated August 29, 2013, the licensee stated, in part, that:

> For compliance with NRC Order EA-12-051 for SFP Level indications, SNC selected Level 1 based on a specific basis as stated in the OIP of February 27, 2013. Level 1 is selected as the current low level system alarm set point for Plant Hatch at elevation 225'-9" for Unit 1 and 226'-2.5" for Unit 2, which is higher than the elevation of the Fuel Pool Cooling (FPC) System Skimmer Surge Tank low level alarm of 222'-0". These surge tanks supply cooling for the FPC pumps for both units. HNP [Hatch] selected Level 1 elevation meets Order EA-12 051 Level 1 criteria.

The NRC staff notes that Level 1 at 225 ft. 9 in., for Unit 1 and 226 ft. 2.5 in., for Unit 2 is adequate for normal SFP cooling system operation. In addition, it is also sufficient for net positive suction head (NPSH) and represents the higher of the two points described above.

In its OIP, the licensee stated that for both Units 1 and 2, Level 2 would be set at elevation 214 ft. 0 in., approximately 10 ft. 0 in. above the highest point of the fuel racks which are at elevation 204 ft. 0 in. The licensee also stated that for both Units 1 and 2, Level 3 would be set at elevation 204 ft. 0 in., which corresponds to the highest point of any rack seated in the SFP.

In its letter dated August 29, 2013, the licensee provided a sketch depicting the SFP elevations identified as Levels 1, 2 and 3, and the SFP level instrumentation measurement range. The NRC staff reviewed this sketch and notes that Level 2 is identified at elevation 214 ft. 0 in. which is approximately 10 ft. above the top of the storage racks. The staff also notes that the licensee designated Level 2 using the first of the two options described in NEI 12-02 for Level 2. Level 3, identified at elevation 204 ft. 0 in. is above the highest point of any spent fuel storage rack seated in the SFP.

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

- Level 1 is the level at which the water height, assuming saturated conditions, is above the elevation where the pumps lose suction from a low level in the skimmer surge tank. It is also sufficient for net positive suction head (NPSH). Hatch Level 1 for both units represents the higher of the two points described in NEI 12-02 for Level 1.

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

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

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

4.2 Evaluation of Design Features

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

4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the primary and backup instrument channel level sensing components would be located and permanently mounted in the SFP, and would use guided wave radar technology. According to the licensee, the primary and backup instrument channels would provide continuous level indication from the high pool level elevation (227 ft. 5 in.) to the top of the spent fuel racks at elevation (204 ft.).

In its letter dated August 29, 2013, the licensee provided a sketch depicting the SFP elevations identified as Levels 1, 2 and 3 and the SFP level instrumentation measurement range. This sketch shows that the level instrument would provide level indication from the high level alarm at 227 ft. 0.5 in. elevation to the top of the spent fuel racks at 204 ft. elevation. In this same letter, the licensee clarified that the elevation for the high pool level elevation was inadvertently referenced as 227 ft. 5 in., and it should have been 227 ft. 0.5 in.

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

The NRC staff concludes that the licensee's design, with respect to the number of channels and measurement range for its SFP, 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 an email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> The fuel pool sensors, located in the spent fuel pool, are connected to the level transmitter, which is located outside the spent fuel pool area, by a coaxial cable. While in the Spent Fuel Pool area, the coax cable will be routed in flex conduit through a cable trench that runs around the fuel pool, the reactor, and the dryer separator pool. The cable trench is 4 in. deep and 8 in. wide. The cable trench

> is recessed 4 in. below the surface of the concrete refuel floor and is covered by a ¼ in. diamond plate steel cover. One side of the cable trench has a 4 in. wide curb that extends 4 in. above the refuel floor. The curb is constructed of steel covered concrete. Each coax cable exits the refuel floor by a penetration to the elevation below. The 1B coax cable routes through the trench to the north end of the Unit 1 Dryer/Separator Pool, then exits the trench and runs in conduit along the north wall approximately 40 ft. to the northwest HVAC duct, where it penetrates through the duct curb to elevation 203 ft. Cabling for the other three level channels (1A, 2A, and 2B) on the refuel floor is routed entirely in the cable trench, with no exposed conduit. Thus the 1A, 2A, and 2B cabling is protected by the recessed trench, the steel cover, and the curb for their entire route on the refuel floor. For both Units 1 and 2, channels A and B are routed separately, as required by NEI 12-02 Section 3.2. After penetrating the refuel floor to elevation 203 ft., all cabling and SFPLIS components including the transmitters and electronics cabinets are protected from damage to the structure over the spent fuel pool.
>
> The spent fuel pool level sensors for Unit 1 are mounted on the north wall of the pool, in opposing corners. Channel 1A is in the northwest corner, channel 1B is in the northeast corner. The north side of the Unit 1 fuel pool is the longest side of the pool; the sensors are separated by approximately 40 ft. (The short side of the Unit 1 fuel pool is 32 ft. 6 in.). The spent fuel pool level sensors for Unit 2 are mounted on the south wall of the pool, in opposing corners. Channel 2A is in the southwest corner, channel 2B is in the southeast corner. The south side of the Unit 2 fuel pool is the longest side of the pool; the sensors are separated by approximately 40 ft. (The short side of the Unit 2 fuel pool is 28 ft. 6-1/2 in.). All other components, including transmitters and electronics cabinets are mounted on elevations below the spent fuel pool floor, and are thus protected against missiles that could result from damage to the structure over the spent fuel pool.

During the onsite audit, the staff walked down the primary and backup SFPLI channels for Units 1 and 2. The NRC staff noted that there appears to be sufficient channel separation within the SFP area between the primary and back-up level instruments, sensor electronics, and routing cables to provide protection against loss of indication of SFP level due to wind-born missiles that may damage the structure over the SFP.

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

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

In an email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> Westinghouse Letter LTR-SEE-II-13-47, "Determination if the Proposed Spent Fuel Pool Level Instrumentation can be Sloshed out of the Spent Fuel Pool during a Seismic Event," Revision 0, dated January 15, 2014, lists the moment force on the mounting bracket due to the sloshing force of the water on the level sensor probe. Hatch calculation CN-PEUS-14-26, "Westinghouse Seismic Load

Calculation," Revision 0, describes the loading on the poolside bracket, and the methodology for calculating the loads, and the acceptance criteria in detail. The loads applied consist of self-weight, dead load of the instrumentation, seismic load, and hydrodynamic load due to the seismic effect (including sloshing). The methodology that was used was generation of GTSTRUDEL models to determine the controlling interaction ratios and bounding forces and moments on the bracket.

Westinghouse document EQ-QR-269, "Design Verification Testing Summary Report for the Spent Fuel Pool Instrumentation," Revision 4, documents the seismic testing for the SFPLIS components, including the transmitters and electronics cabinets. The Westinghouse tests are bounded by the Hatch specific seismic spectra for 1/2 safe shutdown earthquake (SME) and operating-basis earthquake (OBE) spectra.

The maximum loading on the bracket anchors, as determined in CN-PEUS-14-26 is Tmax = 2105.5 lb. and Vmax = 244 lb. (T = Tension and V = Shear). These values include the applied loads from self-weight, dead load of the instrumentation, seismic load, and hydrodynamic load due to the seismic effect (includes the sloshing load discussed in LTR-SEE-II-13-47 Revision 0).

Calculation SCNH-15-001, "SFPLIS Bracket Anchor Calculation," determined the bracket anchorage required to ensure the bracket is correctly mounted in the seismic Category I reactor building. The bracket anchors, four 5/8 in. anchors embedded between 4-1/4 in. and 4-3/4 in., result in an Interaction Coefficient (IC) for expansion anchor of 1.0. This IC of 1.0 is conservative because the combine tension and shear interaction equation uses an exponent of 5/3 consistent with ACI 318-11 Appendix D and Hilti versus an exponent of 2 recommended by Hatch FSAR. Additionally, this is for the worst case scenario which only applies to one anchor. SCNH-15-002," Seismic Mounting Calculation," provides design basis seismic qualification of wall mounted components installed as part of Unit 1 design change package (DCP) SNC549725 and Unit 2 DCP SNC548813. SCNH-15-002 includes analyses of the transmitters, electronics cabinets, coax cable boxes, and junction boxes.

The level sensor probe is attached to a bracket that is mounted to the floor adjacent to the spent fuel pool (Hatch does not use a stilling well.). The bracket consists of a rectangular "Mounting Plate" (13 in. x 14 in.) that is bolted to the refueling floor. A cantilever arm attaches to the center of the base plate and extends over the spent fuel pool. A square (12 in. x 12 in.) "Launch Plate" is attached to the cantilevered arm, and is suspended 22-1/2 in. over the normal water level in the spent fuel pool. The poolside bracket is mounted to the refueling floor using a Hilti Kwik Bolt anchor at each corner of the Mounting Plate.

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

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

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

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

In its OIP, the licensee stated that instrument channel reliability, similar to those applied to the fire protection, will be applied to the components installed in response to this order.

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

4.2.4.2 Instrument Channel Reliability

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

> The instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and/or testing of channel components for the following sets of parameters, as described in the paragraphs below:
>
> - conditions in the area of instrument channel component use for all instrument components,
> - effects of shock and vibration on instrument channel components used during any applicable event for only installed components, and
> - seismic effects on instrument channel components used during and following a potential seismic event for only installed components.

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

The NRC staff reviewed the Westinghouse SFP level instrumentation's qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic. The staff further reviewed the anticipated Hatch's environmental condition during the onsite audit. Below is the staff's assessment of the equipment reliability of Hatch SFP level instrumentation.

4.2.4.2.1 Radiation, Temperature, and Humidity

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> To demonstrate that the SFPLIS equipment will be reliable in BDB conditions, the following calculations and analysis were performed.
>
> Calculation SMNH-15-007, "Radiation Doses for the SFPLIS in a Beyond Design Basis Event in Accordance with NEI 12-02," analyses the radiation doses in both the spent fuel pool area, the transmitter mounting area, and the electronics enclosure mounting area. Models of the Hatch Unit 1 and Unit 2 reactor were developed, and the ORIGEN-ARP and MCNP5 computer codes were used to analyze the radiation doses at the SFPLIS equipment locations.
>
> The Hatch SFPIS uses the Westinghouse supplied 90 degree connector at the spent fuel pool probe, which is qualified to meet the BDB environmental conditions. Calculation DOEJ-DSNC549725-M003, "Room Temperature Evaluation for Spent Fuel Pool Level Instrumentation System (SFPLIS) during Extended Loss of AC Power Event per NEI 12-06," Revision 2.1, was performed to verify the long term temperature of the reactor building during BDB conditions did not exceed the temperature rating of the SFPLIS components. Each area containing equipment was evaluated for long-term temperature effects. The SFPLIS poolside components are designed for humidity conditions of 100% humidity (saturated steam), reference in the Westinghouse WNA-DS-02957-GEN "SFPLI Design Specification." Components outside the spent fuel pool are designed for up to 95% (Non-Condensing) humidity (Reference WNA-DS-02957-GEN Table 4.6-2). Environmental testing of the SFPLIS components was performed, including humidity and temperature tests, and is documented in EQ-QR-269, "Design Verification Testing Summary Report for the SFPIS."
>
> Calculation SMNH-15-007 analyses the radiation doses for the SFPLIS in a BDBEE in accordance with NEI 12-02. For the equipment in the spent fuel pool area, the Level Point 3 (worst case) integrated dose of 8.78 E+06 R is less than the design values for the SFPLIS level probe of 1 E+07 R (Reference WNA-DS-02957-GEN, Table 4.6-1, and SMNH-15-007 Table 2-1). For the equipment outside of the spent fuel pool area, the total integrated dose at the transmitter location is 5.48 E+02 R, and at the electronics cabinet is 5.42 E+02 R (Reference SMNH-15-007 Table 2-1). The Westinghouse design specification for total integrated dose for equipment outside of the spent fuel pool area is 1 E+03 R (Reference WNA-DS-02957-GEN Table 4.6-2).
>
> DOEJ-HDSNC549725-M003 evaluates the temperature in the SFPLIS component areas for an ELAP. The maximum temperature for components in the spent fuel pool area is expected to be 212°F, equal to boiling in the spent fuel pool. The SFPLIS pool area components are designed for the maximum temperature of 212°F. The electronics cabinet equipment, located on Unit 1 elevation 164 ft. and Unit 2 elevation 164 ft. is also rated for 140°F. The long term temperature profile outside of primary containment in the reactor building for elevation 158 ft. area is 118°F. The transmitters, located on Unit 1 elevation 203 ft. and Unit 2 elevation 203 ft. rated design temperature is 140°F. As

> evaluated in DOEJ-HDSNC549725-M003 the temperature in these areas is expected to be less than 140°F. There are no heat sources in the demineralizer hatch access areas where the transmitters are mounted. Therefore, with no local heat sources, the temperature should not exceed the 140°F limit and should be similar to the 118°F established for elevation 158 ft. As documented in DOEJ-HDSNC549725-M003, the equipment meets the requirements of NEI 12-02 for a BDB environmental conditions.

During the onsite audit, the staff reviewed calculations SMNH-15-007 and DOEJ-HDSNC549725-M003 and verified that the SFPLIS is qualified for the environment during a BDB event and within Westinghouse limit specifications.

4.2.4.2.2 Electromagnetic Compatibility

During the onsite audit, the staff requested that the licensee provide an assessment of potential susceptibilities of electromagnetic interference (EMI) and radio frequency interference (RFI) in the areas where the SFPLI is located and explain how to mitigate these susceptibilities. The licensee stated that Westinghouse tested the spent fuel pool level instrumentation system (SFPLIS) for EMI/RFI susceptibility and transmission. The SFPLIS equipment was verified to meet, at minimum, Performance Criterion B (EQ-QR-269Section 4.6.1). The equipment continued to operate after the EMI/RFI interference was removed. The test showed no degradation or loss of function occurred below the performance level specified by the manufacturer. Design change package SNC549725 performed documentation of engineering judgment DOEJ-HDSNC549725-J003 to confirm that the Westinghouse tests met all the requirements of the licensee's standard specification SN9604-002, "Electromagnetic Interference (EMI) Qualification Requirements for Southern Nuclear Power Plant Equipment." The factory integrated functional test (IFT) performed an EMI susceptibility test of the SFPLIS and confirmed that the system recovered to normal operation with no degradation or loss of function after EMI/RFI interference was removed.

The site IFT was based on the factory IFT and was performed at Hatch after implementation of the SFPLIS design change. The site IFT used plant radios to induce EMI/RFI into the SFPLIS at multiple locations. The Hatch SFPLIS was tested at the level sensor probes mounted at the SFP, at points along the coax cable, at the transmitters mounted on elevation 203 ft., at points along the cable from the transmitters to the electronics cabinets, and at each electronics cabinet. The site test confirmed if the system is impacted, and verifed that the SFPLIS recovers to normal operation with no degradation or loss of function after EMI/RFI interference is removed. Any anomalies were addressed by either shielding or use of EMI/RFI exclusion zones. Performance of the site test ensured that the SFPLIS meets the guidance of NEI 12-02 Section 3.4, Qualification, for EMI/RFI effects in the area of instrument channel component use.

Based on the evaluation above, the NRC staff concludes 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

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> Hatch Unit 1 and Unit 2 SFPLIS systems are installed in a manner that an adverse impact from a common cause event is minimized. The A and B channel equipment, cabling, and power supplies are entirely independent for each unit. The A and B cables from the transmitters to the electronics cabinets do not occupy the same conduit, cable tray, or junction box at any point in the cable route. Channel A and B are powered by separate divisions of normal ac power, and each has an independent battery backup.
>
> The level sensors and cabling mounted on the refuel floor in and near the spent fuel pool are physically separated. Separation continues throughout the rest of the SFPLIS system. The level sensors are installed in opposite corners of the pool, separated by a distance greater than the short side of the pool, and the cable from each sensor is routed separately from the sensor to the transmitter. The 1A and 1B transmitters are mounted in the demineralizer hatch access room on elevation 203 ft. in Unit 1, but are on different walls and are separated by more than 20 ft. The 2A and 2B transmitters are mounted in the demineralizer hatch access room on elevation 203 ft. in Unit 2, but are separated by just more than 15 ft. The transmitter cables route in separate conduits or cable trays from the transmitter locations on elevation 203 ft. to the electronics cabinets, located on elevation 164 ft. One channel of electronics cabinet and level display is mounted in the other unit for both Unit 1 and Unit 2. The Unit 1 channel 1A and Unit 2 channel 2A electronics cabinets are mounted in Unit 1 Stair 4 vestibule (Rm 1R208) on elevation 164 ft. The Unit 1 channel 1B and Unit 2 channel 2B electronics cabinets are mounted in Unit 2 Stair 3 vestibule (Rm 2R208) on elevation 164 ft.

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

Based on the evaluation above, the NRC staff concludes 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 <u>Design Features: Power Supplies</u>

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> Electronic cabinets 1A and 2A are mounted in Unit 1 stair 4 vestibule on elevation 164 ft. (Room 1R208). 1A and 2A are normally powered by 120 Vac from panel 1R25-S125. 1R25-S125 is located in and supplied by Unit 1 Division 1 Safety Related 600V Motor Control Center 1R24-S011. During an ELAP, Unit 1 Division 1 is powered by FLEX diesel generator 1, so Channels 1A and 2A are powered by FLEX diesel generator 1. Electronic cabinets 1B and 2B are mounted in Unit 2 stair 3 vestibule on elevation 164 ft. (Room 2R208). 1B and 2B are normally powered by 120 Vac from panel 2R25-S102. 2R25-S102 is located in and supplied by Unit 2 Division 2 safety-related 600 V motor control center 2R24-S012. During an ELAP, Unit 2 Division 2 is powered by FLEX

> diesel generator 2, so Channels 1B and 2B are powered by FLEX diesel generator 2.
>
> As specified in the Westinghouse SFPLIS design specification, the battery in each electronics cabinet must be able to maintain system operation for a minimum of 3 days (WNA-DS-02957, Section 4.3, DS-02957-909). Westinghouse calculation WNA-CN-00300-GEN Revision 0 documents the power consumption for the SFPLIS. Operation of the UPS/Battery system was tested by Westinghouse during design verification testing, and is discussed in EQ-QR-269. Section 5.4.1 is the calculation for the wired configuration used at Hatch. The Hatch SFPLIS battery is calculated to last 4.22 days, or 101.21 hours, providing margin of 1.22 days. The Hatch FLEX procedures require the FLEX diesel generators to be operable after 10 hours (documented in ELAP flowchart 31EO-TSG-003).

During the onsite audit, the staff reviewed drawings H-23350, "Edwin I. Hatch Nuclear Plant Unit No. 2 Master Single Line Diagram," Revision 13; H-23362, "Edwin I. Hatch Nuclear Plant Unit No. 2 Single Line Diagram 600V Bus 2C & 2D," Revision 37; H-27013, "Edwin I. Hatch Nuclear Plant Unit No. 2 Single Line Diagram - Reactor Building 600/208V AC Essential MCC 2B Sheet 1 of 2 MPL 2R24-S012," Revision 43; H-27014, "Edwin I. Hatch Nuclear Plant Unit No. 2 Single Line Diagram - Reactor Building 600/208V AC Essential MCC 2B - Sh. 2 of 2 MPL's 2R24-S012 and 2R25-S102," Revision 39; H-13350, "Edwin I. Hatch Nuclear Plant Unit No. 1 Master Single Line Diagram," Revision 25; H17016, "Edwin I. Hatch Nuclear Plant Unit No. 1 Single Line Diagram Reactor Building 600V AC Essential MCC 1C Sh.1 MPL R24-S011," Revision 33; and H-17010, "Edwin I. Hatch Nuclear Plant Unit No. 1 Single Line Diagram Reactor Building 600 V AC Essential MCC 1C Sh. 1 MPL R24-S011," Revision 42. The staff verified that the electrical ac power supply for the primary and backup SFPLIS are independent so that loss of ac power in one channel will not result in loss of ac power for both channels in the unit. The staff reviewed the battery backup duty cycle during the Westinghouse audit and found it acceptable.

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

4.2.7 Design Features: Accuracy

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> WNA-CN-00301-GEN Revision 2 is the Westinghouse spent fuel pool instrumentation system channel accuracy analysis. The SFPLIS accuracy is not affected by spent fuel pool level. The SFPLIS accuracy is consistent throughout the SFPLIS measurement range. The design accuracy of the Hatch Westinghouse SFPLIS level display, per WNA-CN-00301-GEN (Section 3.1.1) is ±1.60 inch. This accuracy is valid at Level Points 1, 2, and 3. The design accuracy of ±1.60 inch meets the requirements of NEI 12-02 guidance of ± 1 foot at Level Point 2 and Level Point 3.
>
> The Hatch maintenance procedures are in development. They will be similar to the Farley procedures, which are final. During normal operations, a weekly surveillance of the SFPLIS level displays will be performed. The A and B

> channel values will be compared. If the difference between the A and B channel values is greater than 3 inch the Shift Supervisor will be notified to initiate the calibration procedure. Each individual channel is accurate to ±1.60 inch, thus the acceptance criteria is set to 3 inch. The acceptance criteria is less than the NEI 12-02 guidance of ± 1 foot at Level Point 2 and Level Point 3.

The NRC staff concludes that the instrument accuracy of +/- 3 inches is within the accuracy of +/- 1 foot as recommended in NEI 12-02. The licensee will use the manufacturer's design accuracy as acceptance criteria in procedures which will be developed to take corrective action if the accuracy exceed +/- 3 inches.

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

4.2.8 Design Features: Testing

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> A weekly surveillance of the SFPLIS level displays will verify the difference between the A and B level readings. If the reading difference is greater than the acceptance criteria maintenance will be initiated. In addition to calibration initiated from the weekly surveillance, the Hatch SFPLIS will be tested and calibrated within 60 days of a planned refueling outage, as required by NEI 12-02 Section 4.3. The Hatch SFPLIS uses a two position sliding bracket for the level sensor probe mounting bracket. The two position bracket enables each SFPLIS to be lifted 12 in. (equivalent of lowering the water level 12 in.). The bracket enables in-situ testing of the SFPLIS. If additional transmitter calibration is required, a SFPLIS calibration kit is connected to the transmitter on elevation 203 ft. The calibration kit enables calibration of the transmitter independently of the level sensor probe and spent fuel pool water level. The preventive maintenance procedures at Hatch are not finalized. The preventive maintenance procedures will meet systems engineering recommendations and Westinghouse guidance for each SFPLIS component. The planned surveillance interval for Hatch is weekly. If the weekly surveillance determines that one channel is inoperable, the Shift Supervisor will be notified and maintenance actions will be initiated.

The NRC staff noted that the licensee adequately addressed the equipment testing including periodic testing, calibration, and preventive maintenance. These tasks appear to be consistent with the vendor recommendation. The licensee will also perform the channel check to confirm that the two spent fuel pool level instrument channels are reading within the equipment design accuracy.

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

4.2.9 Design Features: Display

In its email dated March 29, 1017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> The location of the Unit 1 and Unit 2 Channel A display panels in stairwell 1R208 is approximately 230 ft. from the Unit 1 and Unit 2 control room. The selected location of the Unit 1 and Unit 2 Channel B display panels in stairwell 2R208 is approximately 250 ft. from the Unit 1 and Unit 2 control room. Using a conservative walking pace of 1 mph, both Unit 1 and Unit 2 A and B display locations each require less than 10 minutes to access the display, perform the local function, and return to the control room. Dose calculation SMNH-15-007 was performed to verify dose rates at the electronics cabinets where the operator reads the displays, in accordance with NEI 12-02 guidance for a BDBEE drain down event. The dose rate at the electronics cabinets is 0.0949 rad/hr at the electronics cabinets confirming the discussion above, that heroic effort is not required due to dose at the electronics cabinets. DOEJ-HDSNC549725-M003 was performed to evaluate the temperature rise at the electronics cabinet location during an ELAP. The temperature profile for elevation 158 ft., after a 24 hour event with no ventilation, for the Unit 1 and Unit 2 reactor building is 118 degrees F. This is a conservative temperature for the stair vestibule where the electronics cabinets are located on elevation 164 ft. There are substantial structures between the SFP and the level displays, and the level displays are mounted at a lower elevation than the floor of the SFP. The shielding provided by the distance and concrete between the level displays and the SFP, combined with the short transit duration from the control room to the level display location, means heroic effort is not required for operators to monitor the SFP level at any level point, including Level Point 3. The display location remains habitable considering the minimal time required to access the displays, distance from the SFP, presence of substantial intervening structures, and the lack of heat producing equipment within the room during accident conditions. Both locations are expected to allow for prompt, non-heroic access to the displays from the control room. SNC intends to periodically monitor the display at one to two hour intervals during accident conditions. During normal operation a weekly surveillance of the SFPLIS will be performed.

During the onsite audit, the staff walked down the display locations. The staff also reviewed Calculation SMNH-15-007, "Radiation Dose to Spent Fuel Pool Level Instrumentation System in Accordance with NEI 12-02," Revision 2, and DOEJ-HDSNC549725-M003, "Room Temperature Evaluation for Spent Fuel Pool Level Instrumentation System (SFPLIS) during Extended Loss of AC Power Event per NEI 12-06." The staff concluded that the display locations are promptly accessible and habitable during a BDBEE.

Based on the evaluation above, the NRC staff concludes 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 SFP instrumentation.

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated, in part, that:

> A systematic approach will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training. Personnel will complete training prior to being assigned responsibilities associated with this instrument.

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

4.3.2 Programmatic Controls: Procedures

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> TE 939638 has been written to track development of the Hatch Procedures. Procedures will be developed to ensure the following objectives. The Hatch procedures will be developed using Vogtle and Farley SFPLIS procedures as a guide, and will be consistent with Westinghouse recommendations for maintenance, testing, calibration, and inspection.
>
> Procedures will be developed to ensure the following objectives:
>
> a) System Inspection: Verify system components are in place, complete, and configured correctly.
> b) Calibration and Test: Verify the system is within the specified accuracy, functioning as designed, and indicating SFP water level.
> c) Maintenance: Establish and define maintenance requirements (both scheduled and preventative) and activities necessary to minimize possibility of system interruption.
> d) Repair: Specify steps for problem identification, repair, and replacement activities in the event of system malfunction.
> e) Operation: Provide sufficient instructions for operation by the plant operations staff.
> f) Responses: Define actions to be taken upon observation of system level indications, including actions to be taken at the levels defined in NEI 12-02.

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

4.3.3 Programmatic Controls: Testing and Calibration

In its email dated March 29, 2017 (ADAMS Accession No. ML17090A163), the licensee documented the response from the onsite audit and stated that:

> A weekly surveillance of the SFPLIS level displays will verify the difference between the A and B level readings. If the reading difference is greater than the acceptance criteria calibration will be initiated. In addition to calibration initiated from the weekly surveillance, the Hatch SFPLIS will be tested and calibrated within 60 days of a planned refueling outage, as required by NEI 12-02 Section 4.3. Preventive maintenance procedures to include tests, inspection, and periodic replacement of the backup batteries will be developed based on Westinghouse recommendations and system engineering input (based on existing procedures at Plant Farley and Vogtle).
>
> If both channels are out of service, a condition report will be initiated and addressed through SNC's corrective action program and action tracking application. Provisions associated with out of service (OOS) or non-functional equipment, including allowed outage times and compensatory actions, will be consistent with the guidance provided in Section 4.3 of NEI 12-02. If one OOS channel cannot be restored to service within 90 days, appropriate compensatory actions, including the use of alternate suitable equipment, will be taken. If both channels become OOS, actions would be initiated within 24 hours to restore one of the channels to operable status and implement appropriate compensatory actions, including the use of alternate suitable equipment and/or supplemental personnel, within 72 hours.
>
> Southern Company is implementing the Westinghouse SFPLIS at Hatch, Farley and Vogtle. The systems are standardized, and use the same electronics components, cable types, and coax cable lengths. Spare components will be maintained by SNC to ensure that an out of service SFPLIS channel can be returned to service within the 90 days required by NEI 12-02 Section 4.3.
>
> If both channels are OOS a condition report will be initiated and addressed through SNC's corrective action program, and action tracking application. SNC will maintain sufficient spare parts for the SFPLIS to provide assurance that a channel can be restored to service within 90 days. In the event that one channel could not be restored within 90 days, the corrective action program would monitor the condition and ensure that effort and notification was elevated as needed.

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

4.4 Conclusions for Order EA-12-051

In its letter dated February 13, 2017 (ADAMS Accession No. ML17044A414), the licensee stated that it met 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's plans conform to the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. Based on the evaluation above, the NRC staff concludes that if the SFP level instrumentation is installed at Hatch 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 October 2015 (ADAMS Accession No. ML15349A801). The licensee reached its final compliance date on February 13, 2017, and has declared that both of the reactors are in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs that if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders.

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EDWIN I. HATCH NUCLEAR 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
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