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

August 30, 2016

Mr. Benjamin C. Waldrep
Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
New Hill, NC 27562-9300

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – 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. MF0874 AND MF0792)

Dear Mr. Waldrep:

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

By letter dated February 28, 2013 (ADAMS Accession No. ML13112A020), Duke Energy Progress, Inc. (Duke, the licensee), submitted its OIP for Shearon Harris Nuclear Power Plant, Unit 1 (Harris, HNP) in response to Order EA-12-049. At six month intervals following the submittal of the OIP, Duke 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 (SE). By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all operating power licensees and construction permit holders that the NRC 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 12, 2014 (ADAMS Accession No. ML13364A214), and April 14, 2015 (ADAMS Accession No. ML15083A024), the NRC issued an Interim Staff Evaluation (ISE) and an audit report, respectively on the licensee's progress.

By letter dated July 10, 2015 (non-public), as supplemented by letter dated June 29, 2016 (ADAMS Accession No. ML16182A047), Duke submitted its compliance letter and Final Integrated Plan in response to Order EA-12-049. The compliance letters stated that Duke had achieved full compliance with Order EA-12-049.

By letter dated February 28, 2013 (ADAMS Accession No. ML13086A096), Duke submitted its OIP for HNP in response to Order EA-12-051. At six month intervals following the submittal of the OIP, Duke 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 SE. By letters dated November 19, 2013 (ADAMS Accession No. ML13280A482), and April 14, 2015 (ADAMS Accession No. ML15083A024), the NRC issued an ISE and audit report on Duke's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all operating power licensees and construction permit holders that the NRC staff is conducting in-office and on-site 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 July 10, 2015 (non-public) Duke submitted its compliance letter and FIP in response to Order EA-12-051. The compliance letter stated that Duke had achieved full compliance with Order EA-12-051.

The enclosed SE provides the results of the NRC staff's review of Duke's strategies for HNP. The intent of the SE is to inform Duke whether its integrated plans, if implemented as described, will adequately address the requirements for compliance with Orders EA-12-049 and EA-12-051. The NRC staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515/191, "Inspection of the Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans," Revision 1 (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 Stephen Monarque, Orders Management Branch, HNP Project Manager, at 301-415-1544 or at Stephen.Monarque@nrc.gov.

Sincerely,

Mandy K Halter

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

Docket No.: 50-400

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

TABLE OF CONTENTS

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

DUKE ENERGY PROGRESS, INC.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

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

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

Enclosure

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC regulations and processes and determining if the agency should make additional improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 [Reference 1]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

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

2.1 Order EA-12-049

Order EA-12-049, Attachment 2, [Reference 4] requires that operating power reactor licensees and construction permit holders use a three-phase approach for mitigating BDBEE. 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.

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

On August 21, 2012, following several submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0 [Reference 6] to the NRC to provide specifications for an industry-developed methodology for the development, implementation, and maintenance of guidance and strategies in response to the Order EA-12-049. The NRC staff reviewed NEI 12-06 and on August 29, 2012, issued Japan Lessons-Learned Directorate (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" [Reference 7], endorsing NEI 12-06, Revision 0 [Reference 6], with clarifications, as an acceptable means of meeting the requirements of Order EA-12-049. On September 7, 2012, the NRC staff published a notice of the availability of JLD-ISG-2012-01 in the *Federal Register* (77 FR 55230).

2.2 Order EA-12-051

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

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

structure over the spent fuel pool. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

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

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

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

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

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

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

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

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

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

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

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

On August 24, 2012, following several NEI submittals and discussions in public meetings with NRC staff, the NEI submitted document NEI 12-02, "Industry Guidance for Compliance With NRC Order EA-12-051, To Modify Licenses With Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1 [Reference 8] to the NRC to provide specifications for an industry-developed methodology for compliance with Order EA-12-051. On August 29, 2012, the NRC staff issued its final version of JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation" [Reference 9] endorsing NEI 12-02, Revision 1 [Reference 8], as an acceptable means of meeting the requirements of Order EA-12-051 with certain clarifications and exceptions. On September 7, 2012, the NRC staff published a notice of the availability of JLD-ISG-2012-01 in the *Federal Register* (77 FR 55230).

3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 [Reference 10], Duke Energy Progress, Inc. (Duke, the licensee) submitted its Overall Integrated Plan (OIP) for Shearon Harris Nuclear Power Plant, Unit 1 (Harris, HNP) in response to Order EA-12-049. By letters dated August 27, 2013, February 27, 2014, August 25, 2014, and February 23, 2015 [References 11, 12, 13, and 14, respectively] the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 15], the NRC notified all operating power licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 64]. By letters dated February 12, 2014 [Reference 16], and April 14, 2015 [Reference 17], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated July 10, 2015 [Reference 18], as supplemented by letter dated June 29, 2016 [Reference 94], 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 from one phase to the next 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 loss of normal access to the ultimate heat sink (LUHS). Thus, the ELAP with LUHS is used as a surrogate for a BDBEE. The initial conditions and assumptions for the analyses are stated in NEI 12-06 [Reference 6], Section 3.2.1, and include the following:

1. The reactor is assumed to have safely shutdown 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.

Harris is a three-loop Westinghouse pressurized-water reactor (PWR) with a dry ambient pressure containment. The FIP describes Duke's three-phase approach to mitigate a postulated ELAP event.

At the onset of an ELAP, the operators initiate reactor and turbine trip and isolate the reactor coolant system (RCS) to prevent inventory loss. Duke will use the steam generators (SGs) as the heat sink for core cooling. The SGs will be fed by the Turbine-Driven Auxiliary Feedwater (TDAFW) pump, which is capable of operating for a minimum of 24 hours during an ELAP event. The condensate storage tank (CST) is the initial water source to the TDAFW pump. Duke's core cooling strategy involves depressurization of the SGs, which will be performed using the SG Power Operated Relief Valves (PORVs). The SGs will be depressurized to maintain cool down of the RCS cold legs at a rate of approximately 100 degrees farenheit per hour (°F)/hr). The RCS cold leg temperature will be less than 450°F within 4 hours of the start of the event. The RCS cold leg temperature and pressure will be less than 350°F and 400 pounds per square inch gage (psig), respectively, within 24 hours. The target SG pressure is 230 psig if the cold leg accumulators (CLAs) have not been isolated and 160 psig if the CLA have been isolated.

The Phase 2 core cooling strategy continues to use the SGs as the heat sink. Duke will transition to an electric motor-driven FLEX Auxiliary Feedwater (AFW) Pump, which can deliver water to the SG injection point at 325 gallons per minute (gpm) and 363 psig. Within 24 hours of the ELAP event initiation, the target SG pressure is 120 psig, and the target RCS temperature is 350°F.

The CST water supply is can provide at least 36 hours of cooling water for non-seismic events. For a seismic event, the CST is credited with providing at least 21 hours of cooling water. As CST depletion approaches, an Emergency Service Water (ESW) header may be pressurized by one of two portable FLEX ESW Pumps taking suction from an ESW pump bay in the ESW and Cooling Tower Make-up Water Intake Structure. Duke will have an ESW header pressurized and aligned to the AFW system within 36 hours of the start of the event. If CST depletion is anticipated to occur prior to ESW header pressurization, Duke will use inventory directly from the Reactor Makeup Water Storage Tank (RMWST) and the Refueling Water Storage Tank (RWST) for AFW until ESW is pressurized and aligned to AFW.

Duke credits the use of the Auxiliary Reservoir if all other sources are exhausted or unavailable. If higher quality water sources are available from non-credited sources (e.g., tanks not protected from applicable extreme external hazards), Duke would use those sources first. Specifically, in order of preference, Duke would refill the CST using water from the Demineralized Water Storage Tank (DWST), Hotwell, or the Pretreated Water Storage Tank (PWST) instead of using water from the Auxiliary Reservoir.

Duke will restore power to selected plant equipment at HNP and provide power to FLEX equipment using a permanently pre-staged FLEX Diesel Generator (DG) rated at 480V located in the Emergency Diesel Generator (EDG) Building Bay 2B. A second permanently pre-staged DG provides backup capability. Duke plans to establish FLEX Power within 6 hours of the event.

An electric-powered FLEX RCS Pump will be mobilized at the Charging/Safety Injection Pump (CSIP) rooms to provide RCS makeup as well as maintain subcriticality, using RCS boration. The sources of borated water will be from the Boric Acid Tank (BAT), Alternate Seal Injection (ASI) Tank, or RWST. The RCS makeup will be initiated 10.5 hours into the event. After boration is complete, the CLAs will be isolated in preparation for RCS cooldown below 370°F cold-leg temperature (160 psig SG pressure), as this will prevent the injection of nitrogen gas into the RCS.

The Phase 3 core cooling strategy continues to use the SGs as the heat sink. If necessary, the FLEX RCS, AFW, and ESW pumps can be replaced by pumps from the National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC). Duke will also receive water treatment equipment from the NSRC to ensure a long-term source of clean water for core cooling. The RWST can be resupplied by several methods. Borated makeup water can be mixed in the in the Recycle Monitor Batch Tank for transfer to the RWST. Inventory from the SFPs can be transferred by gravity feed to the RWST as well.

3.2 Reactor Core Cooling Strategies

The guidance contained in NEI 12-06 provides for licensees to maintain or restore cooling to the reactor core in the event of an ELAP concurrent with a LUHS. 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 [Reference 6], Phase 1 of the licensee's core cooling strategy credits installed equipment (other than that presumed lost due to the ELAP and LUHS) that is considered robust to natural hazards of design-basis magnitude, in accordance with the guidance in NEI 12-06 [Reference 6]. In Phase 2, robust installed equipment is supplemented by onsite FLEX equipment, which is used to cool the core either directly (e.g., with pumps and hoses) or indirectly (e.g., through the use of 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.

To adequately cool the reactor core under ELAP conditions, two fundamental physical requirements exist: (1) a heat sink is necessary to accept the heat transferred from the reactor core to coolant in the RCS and (2) sufficient RCS inventory is necessary to transport heat from the reactor core. Furthermore, inasmuch as heat removal requirements for the ELAP event consider only residual heat, the RCS inventory should be replenished with borated coolant in order to maintain the reactor in a subcritical condition as the RCS is cooled and depressurized.

As reviewed in this section, the licensee's core cooling analysis for the ELAP and LUHS event presumes that, per endorsed guidance from NEI 12-06 [Reference 6], Duke would have been operating at full power prior to the event. Therefore, the SGs may be credited as the heat sink for core cooling during the ELAP and LUHS event. Maintenance of sufficient RCS inventory, despite ongoing system leakage expected under ELAP conditions, is accomplished through a combination of installed systems and FLEX equipment. The specific means used by the licensee to accomplish adequate core cooling during the ELAP and LUHS event are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions in which the plant is shut down or being refueled is reviewed separately in Section 3.11 of this evaluation.

3.2.1 Core Cooling Strategy and RCS Makeup for Non-Seismic Event

3.2.1.1 Core Cooling Strategy

3.2.1.1.1 Phase 1

As stated in Duke's FIP [Reference 18], the heat sink for core cooling in Phase 1 is provided by the three SGs, which are fed simultaneously by the TDAFW pump with inventory supplied by the CST, which has a volume of 415,000 gallons. For a non-seismic event, the credited CST capacity is 80 percent (324,800 gallons) which would be sufficient for at least 36 hours. Because the non-seismic condensate transfer pump suction line nozzle could fail during a seismic event and partially drain the tank, the available CST volume following a seismic event, is limited to 238,000 gallons (21 hours coping time). There are additional protected alternate water sources available to ensure an adequate feedwater supply is maintained in support of mitigation strategies following a seismic event. The RWST has a creditable volume of 407,000 gallons, and the seismically-protected RMWST has an available credited volume of approximately 28,500 gallons.

Duke's Phase 1 strategy directs the initiation of a cooldown and depressurization of the RCS within two hours of the initiation of the ELAP and LUHS event. Using the SG PORVs to remove decay heat by relieving steam, Duke would cool down the RCS from post-trip conditions at a rate of nearly 100°F/hr. This results in the RCS cold leg temperature being less than 450°F within 4 hours of the event initiation. Cooldown and depressurization of the RCS significantly extends the expected coping time under ELAP and LUHS conditions because it reduces the RCS leakage and it reduces the potential for damage to the RCP seals.

The RCS inventory tends to gradually diminish. As is typical of operating PWRs, HNP does not have installed ac-independent primary makeup pumps. Therefore, the licensee has performed an analysis and concluded that the CLAs will maintain sufficient RCS inventory in Phase 1 to support heat transfer to the SGs via natural circulation.

3.2.1.1.2 Phase 2

Duke's FIP states that the primary strategy for core cooling in Phase 2 is to continue using the SGs as a heat sink. Duke will establish plant conditions at HNP to allow transition to an electric motor-driven FLEX AFW pump, both to provide defense-in-depth for maintaining an adequate heat sink and to allow the securing of the TDAFW pump. Duke will transition to the FLEX AFW pump prior to 24 hours after initiation of the ELAP event. This electric motor driven pump will be powered by an installed FLEX electrical distribution system, supported by one of two FLEX 480V diesel generators.

In addition to powering the FLEX AFW pump, this FLEX distribution system provides power for multiple Phase 2 loads including the FLEX RCS pumps, SG PORV hydraulic pump motors, and alternate power to Motor Control Center (MCC) buckets for valve operation. FLEX power will be established within 6 hours of the event, and the FLEX AFW pump will be aligned within 12 hours.

According to Duke's calculations, the CST is capable of supplying the SGs for at least 36 hours following a non-seismic event. To provide an essentially unlimited source of secondary makeup in Phase 2, Duke stated that a portable diesel-driven FLEX pump would be deployed at the ESW pump bay in the Intake Structure. The ESW pump bay is gravity fed with water through existing ESW piping from the Screening Structure. If necessary, the traveling screens can be bypassed by the pump taking suction directly from the Auxiliary Reservoir Intake Canal and discharging to the applicable ESW header. The ESW piping could in turn be aligned to provide suction to the electric motor driven FLEX pump, when the CST inventory is depleted. Thus, the Auxiliary Reservoir will provide a sustained water supply to the SGs.

3.2.1.1.3 Phase 3

The Phase 3 strategy continues to use the SGs as the heat sink. Equipment from the NSRC, such as water treatment equipment and replacement pumps, will be available if needed to ensure long-term core cooling.

3.2.1.2 RCS Makeup Strategy

3.2.1.2.1 Phase 1

Throughout the ELAP event, RCS inventory would gradually diminish due to leakage through RCP seals and other leakage points in the RCS. As is typical of operating PWRs, HNP does not have installed ac-independent primary makeup capability. However, passive injection from the CLAs to the RCS will occur, during Phase 1, which will offset the contraction of the RCS inventory that occurs due to the RCS cooldown as well inventory loss from RCS leakage. According to the licensee's calculations, adequate shutdown margin is available in Phase 1 due primarily to the control rods and the presence of xenon-135, even if boration from the accumulators is not credited.

3.2.1.2.2 Phase 2

In order to maintain sufficient RCS inventory in Phase 2 to support natural circulation, Duke stated that one of two portable electric motor-driven high-pressure FLEX pumps, powered by the FLEX electrical distribution network, will supply highly borated water from the BAT to the RCS. The boron will ensure long-term subcriticality. Duke has calculated that 4,800 gallons of 7,000 parts per million (ppm) borated water from the BAT is needed to prevent a return to criticality in the most-limiting scenario, and the BAT has a minimum usable volume of 24,150 gallons. FLEX RCS makeup will be initiated 10.5 hours into the event. Initiating makeup at this time provides margin to both (1) the time of 20 hours into the event Duke calculated as conservatively preventing a return to criticality; and (2) the time of 14.8 hours into the ELAP event at which Duke calculated that HNP would enter reflux cooling. The FLEX RCS pump's suction and discharge will be aligned to connection points on the suction and discharge headers for the CSIPs, respectively. The deployment and alignment of a FLEX RCS pump will allow Duke to use the BAT as the initial source of makeup to the RCS. The RWST is credited as the long-term source of makeup to replenish the RCS inventory. The RWST has a creditable volume of 407,000 gallons in most scenarios but is limited to approximately 187,000 gallons for a tornado missile event. If available, additional BAT inventory may be diluted and used for long-term RCS inventory management.

The CLAs will be isolated prior to depressurizing the SGs to 160 psig to preclude nitrogen gas entering the RCS and interfering with natural circulation. The power to the CLA outlet isolation motor operator valves would be supplied by the FLEX distribution network in Phase 2.

3.2.1.2.3 Phase 3

In Phase 3, equipment from the NSRC, such as water treatment equipment and replacement pumps, will be available to ensure long-term core cooling. The RWST inventory can be replenished by diluting the high boron concentration in the BAT, Alternate Seal Injection (ASI) Tank, or Recycle Monitor Tank with the effluent from the NSRC water treatment equipment.

3.2.2 Variations to Core Cooling Strategy for Seismic Event

For a seismic event, the CST is credited to be available for only 21 hours, due to a non-seismic condensate transfer pump suction line nozzle. This nozzle is assumed to fail and reduce the amount of water available for core cooling. If CST depletion is anticipated to occur before the operators align the FLEX pump and pressurize the ESW header (which the FIP states will occur within 36 hours of the start of the ELAP event), Duke will use inventory directly from the RMWST and the RWST for AFW. The licensee will switch the AFW suction to the ESW once it is pressurized. The available credited volume in the RWST and RMWST is sufficient to supply AFW until an ESW header is available. Section 3.10.1 contains more detailed discussion related to CST depletion.

The ASI Tank, Recycle Monitor Tank, and Recycle Monitor Batch Tanks are not seismically robust. They can be used for RCS inventory control and makeup to the RWST if available, but the core cooling strategy does not depend on them.

3.2.3 Staff Evaluations

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

Guidance document NEI 12-06 [Reference 6] provides guidance that the baseline assumptions have been established on the presumption that other than the LUHS, 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

Core Cooling – Phase 1

Duke's Phase 1 core cooling FLEX strategies rely on the existing TDAFW pump to remove heat from the RCS by providing cooling water to the three SGs. As described in Harris Updated Safety Analysis Report (UFSAR) Table 3.2.1-1, the TDAFW pump and the AFW System pump and piping are safety-related, seismic Category I and, with the exception of the CST, are located in the Reactor Auxiliary Building (RAB). The TDAFW pump control power is provided to the control room by the safety-related, 125V Class 1E Battery 1B-SB. Emergency Operating Procedure (EOP)-ECA-0.0, "Loss of All AC Power," [Reference 40] directs the operators to use local manual operation of the TDAFW pump if control power is not available. The licensee's Phase 1 core cooling FLEX strategies relies on the CST as the water source for the TDAFW. Further explanation and the NRC staff's evaluation of the robustness and availability of water sources for an ELAP event is discussed in Section 3.10 of this safety evaluation (SE).

Additionally, Duke plans to depressurize the SGs using the SG PORVs. As described in Harris UFSAR Table 3.2.1-1, the PORVs are safety-related, the piping up to and including the valves is seismic Category I, and these components are located in the RAB. As described in the licensee's FIP, the PORVs can be controlled from the Main Control Room (MCR) with power provided by the Class 1E safety-related batteries during Phase 1 and the FLEX DG during Phase 2. As detailed in the FIP, the PORVs are hydraulically operated and have accumulators designed to support a minimum of four full valves open and close cycles. Procedure EOP-ECA-0.0 [Reference 40], directs the operators to recharge the accumulators via either hand pumps located near the valves or using the installed electrical pumps once repowered from the FLEX DG.

As described in Harris UFSAR Table 3.2.1-1 and 3.5.1-1 the RAB is a seismic Category I structure that is designed to withstand design-basis wind/tornado loadings and missile impacts. Furthermore, the RAB is located above the probable maximum flood height as described in Harris UFSAR Table 3.4.1-1. The NRC staff noted that the TDAFW pumps are located in a temperature-controlled area of the RAB. Further explanantion and the NRC staff's evaluation of equipment operation during an ELAP event will be addressed in Section 3.9 of this SE. Based on the location of the TDAFW pump and PORVs, both systems should be available during an ELAP event.

Core Cooling – Phase 2

Duke's Phase 2 core cooling strategy continues to use the SGs as the heat sink. Harris will transition to an electric motor-driven FLEX AFW Pump, with water supplied from the auxiliary reservoir using the FLEX ESW Pump, and does not rely on any installed plant SSCs other than installed systems with FLEX connection points and water sources discussed in Sections 3.7 and 3.10, respectively.

Core Cooling – Phase 3

Duke's Phase 3 core cooling strategy for HNP does not rely on any additional installed plant SSCs other than those discussed in Phase 2.

RCS Inventory Control – Phase 1

Duke calculated that no action is required for RCS inventory control in Phase 1. Initiation of active injection of borated water into the RCS will not be required to begin until 10.5 hours into the analyzed ELAP event. This initiation time for RCS injection would satisfy criteria for both maintaining natural circulation flow in the RCS and ensuring the reactor has adequate shutdown margin.

RCS Inventory Control – Phase 2

Duke's Phase 2 RCS inventory strategy for HNP will use an electric motor-driven FLEX RCS Pump and does not rely on any installed plant SSCs other than installed systems with FLEX connection points and borated water sources discussed in Sections 3.7 and 3.10, respectively.

RCS Inventory Control – Phase 3

Duke's Phase 3 RCS inventory strategy for HNP does not rely on any additional installed plant SSCs other than those discussed in Phase 2.

Based on the location and design of the credited plant SSCs, as described in Duke's UFSAR, and if implemented according to Duke's control strategy, as described in the FIP, the credited plant SSCs should be available to support core cooling during an ELAP, consistent with NEI 12-06, Section 3.2.1.3, Condition 6.

3.2.3.1.2 Plant Instrumentation

According to Duke's FIP, the following instrumentation would be relied upon to support the licensee's core cooling and RCS inventory control strategy:

- RCS hot leg temperature
- RCS cold leg temperature
- RCS wide range pressure
- SG narrow range level
- SG wide range level

- Core exit thermocouple temperature
- Pressurizer level
- Reactor Vessel Level Indicating System (RVLIS)
- [Turbine or Motor] AFW pump flow
- SG pressure
- CST level
- Battery capacity / dc bus voltage
- Safety-related battery charger voltage
- Safety-related battery charger amperage
- Neutron flux
- RWST level
- RMWST level
- BAT level

Duke's FIP states that, as recommended by Section 5.3.3 of NEI 12-06 [Reference 6], procedures have been developed to read the above instrumentation locally using portable instruments or other alternate means, where applicable. The alternate means of obtaining key parameters are as follows:

- Obtain reading at Process Instrumentation Cabinets (RCS temperatures)
- Obtain reading at transmitter in RAB (RCS pressure and BAT level)
- Obtain reading at containment penetration (SG levels, core exit thermocouple temperature, pressurizer level)
- Re-power from FLEX electrical distribution system (RVLIS, neutron flux)
- Obtain local gage reading ([FLEX or Turbine] AFW pump flow, CST level - convert suction pressure to elevation head, RWST level, RMWST level)
- Obtain local temporary gage reading in steam tunnel (SG pressure)
- Obtain reading at battery (battery voltage)

As described in Duke's FIP, and in accordance with NEI 12-06 [Reference 6], Section 5.3.3.1, guidelines for obtaining critical parameters locally are provided in plant procedures. Portable FLEX equipment is supplied with local instrumentation needed to operate the equipment. These procedures include the use of the FLEX equipment and instrumentation.

3.2.3.2 Thermal-Hydraulic Analyses

Duke concluded that its mitigating strategy for reactor core cooling would be adequate based, in part, on a generic thermal-hydraulic analysis performed for a reference Westinghouse four-loop reactor using the NOTRUMP computer code. The NOTRUMP code and corresponding evaluation model were originally submitted in the early 1980s as a method for performing licensing-basis safety analyses of small-break loss-of-coolant accidents (LOCAs) for Westinghouse PWRs. Although NOTRUMP has been approved for performing small-break LOCA analysis under the conservative Appendix K paradigm and constitutes the current evaluation model of record for many operating PWRs, the NRC staff had not previously examined its technical adequacy for performing best-estimate simulations of the ELAP event. Therefore, in support of mitigating strategy reviews to assess compliance with Order EA-12-049, the NRC staff evaluated licensees' thermal-hydraulic analyses, including a limited review of the significant assumptions and modeling capabilities of NOTRUMP and other thermal-hydraulic codes used for these analyses. The NRC staff's review included performing confirmatory analyses with the TRACE code to obtain an independent assessment of the duration that reference reactor designs could cope with an ELAP event prior to providing makeup to the RCS.

Based on its review, the NRC staff questioned whether NOTRUMP and other codes used to analyze ELAP scenarios for PWRs would provide reliable coping time predictions in the reflux or boiler-condenser cooling phase of the event because of challenges associated with modeling complex phenomena that could occur in this phase, including boric acid dilution in the intermediate leg loop seals, two-phase leakage through RCP seals, and primary-to-secondary heat transfer with two-phase flow in the RCS. Due to the challenge of resolving these issues within the compliance schedule specified in Order EA-12-049, the NRC staff requested that industry provide makeup to the RCS prior to entering the reflux or boiler-condenser cooling phase of an ELAP, such that reliance on thermal-hydraulic code predictions during this phase of the event would not be necessary.

Accordingly, the ELAP coping time prior to providing makeup to the RCS is limited to the duration over which the flow in the RCS remains in natural circulation, prior to the point where continued inventory loss results in a transition to the reflux or boiler-condenser cooling mode. In particular, for PWRs with inverted U-tube SGs, the reflux cooling mode is said to exist when vapor boiled off from the reactor core flows out the saturated, stratified hot leg and condenses on SG tubes, with the majority of the condensate subsequently draining back into the reactor vessel in countercurrent fashion. Quantitatively, as reflected in documents such as Pressurized-Water Reactor Owners Group (PWROG) PWROG-14064-P, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances," Revision 0 [Reference 65] industry has proposed defining this coping time as the point at which the one-hour centered time-average of the flow quality passing over the SG tubes' U-bend exceeds one-tenth (0.1). As discussed further in Section 3.2.3.4 of this evaluation, a second metric for ensuring adequate coping time is associated with maintaining sufficient natural circulation flow in the RCS to support adequate mixing of boric acid.

With specific regard to NOTRUMP, preliminary results from the NRC staff's independent confirmatory analysis performed with the TRACE code indicated that the coping time for Westinghouse PWRs under ELAP conditions could be shorter than predicted in WCAP-17601-P, "Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs," Revision 1 [Reference 96]. Subsequently, a series of additional simulations performed by the NRC staff and Westinghouse identified that the discrepancy in predicted coping time could be attributed largely to differences in the modeling of RCP seal leakage. (The topic of RCP seal leakage will be discussed in greater detail in Section 3.2.3.3 of this SE.) These comparative simulations showed that when similar RCP seal leakage boundary conditions were applied, the coping time predictions of TRACE and NOTRUMP were in adequate agreement. From these simulations, as supplemented by review of key code models, the NRC staff obtained sufficient confidence that the NOTRUMP code may be used in conjunction with the WCAP-17601-P evaluation model for performing best-estimate simulations of ELAP coping time prior to reaching the reflux cooling mode.

Although the NRC staff obtained confidence that the NOTRUMP code is capable of performing best-estimate ELAP simulations prior to the initiation of reflux cooling using the one-tenth flow-quality criterion discussed above, the NRC staff was unable to conclude that the generic analysis performed in WCAP-17601-P [Reference 96] could be directly applied to all Westinghouse PWRs, as the vendor originally intended. In PWROG-14064-P [Reference 65] and PWROG-14027-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power, Task 3: Evaluations of Revised Seal Flow Rate on Time to Enter Reflux Cooling and Time at which the Core Uncovers," Revision 3 [Reference 38], the industry subsequently recognized that the generic analysis would need to be scaled to account for plant-specific variation in RCP seal leakage and potentially other plant parameters. Harris was the reference case for the three-loop (T-cold upper head) Westinghouse plant in PWROG-14064-P [Reference 65], and no significant differences exist between the plant parameters applicable to HNP and those assumed in the reference case analysis. Several differences exist between Duke's planned mitigating strategy and that assumed in the reference case, including in particular the termination points for the RCS cooldown. The NRC staff has reviewed these differences and concludes that the reduced RCS pressures associated with Duke's planned mitigating strategy would tend to provide additional conservative margin to the generically calculated coping time of 14.8 hours determined in PWROG-14027-P, Revision 3 [Reference 38].

Therefore, based on the evaluation above, the NRC staff concludes that the licensee's analytical approach is sufficiently accurate for determining the sequence of events, including the time-sensitive operator actions, and the required equipment to mitigate the ELAP event.

3.2.3.3 Reactor Coolant Pump (RCP) Seals

Leakage from RCP seals is among the most significant factors in determining the duration that a pressurized-water reactor can cope with an ELAP event prior to initiating RCS makeup. An ELAP event would interrupt cooling to the RCP seals, resulting in increased leakage for Westinghouse standard seal packages and the potential for failure of elastomeric o-rings and other components, which could further increase the leakage rate. As discussed above, as long as adequate inventory is maintained in the RCS, natural circulation can effectively transfer residual heat from the reactor core to the SGs and limit local imbalances in boric acid concentration. Along with cooldown-induced shrinkage of the RCS inventory, cumulative leakage from RCP seals governs the duration over which natural circulation can be maintained in the RCS. Furthermore, the seal leakage rate at the depressurized condition can be a controlling factor in determining the flow capacity requirement for FLEX pumps to offset ongoing RCS leakage and recover adequate system inventory.

The Model 93A RCPs at Harris use a three-stage Westinghouse 93ACS seal package. As noted in Section 3.2.3.2, Duke is relying on thermal-hydraulic analysis performed with the NOTRUMP code, as documented in WCAP-17601-P [Reference 96] and PWROG-14064-P [Reference 65], to determine the time at which makeup would be required to maintain adequate natural circulation flow in the RCS. In accordance with loss-of-seal-cooling analysis and testing performed in the 1980s and documented in WCAP-10541-P, "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Revision 2 [Reference 66], the ELAP analysis in WCAP-17601-P and PWROG-14064-P [Reference 65] assumed a leakage rate at nominal post-trip cold leg conditions (i.e., 2,250 pounds per square inch absolute (psia) and 550°F) of 21 gpm for each RCP, plus an additional 1 gpm of operational leakage. In the WCAP-17601-P [Reference 96] and PWROG-14064-P [Reference 65] analysis, both seal and operational leakage were assumed to vary according to the critical flow correlation modeled in the NOTRUMP code as the reactor was cooled down and depressurized.

Recent assessments of RCP seal leakage behavior under ELAP conditions by industry analysts and NRC staff identified several issues with the original treatment of seal leakage from standard Westinghouse seal packages. Key concerns are documented in the Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-14-1, dated February 10, 2014 [Reference 67], including (1) the initial post-trip leakage rate of 21 gpm does not apply to all Westinghouse PWRs due to variation in seal leakoff line hydraulic configurations, (2) seal leakage does not appear to decrease with pressure as rapidly as predicted by the analysis in WCAP-17601-P, and (3) some reactors may experience post-trip cold leg temperatures in excess of 550°F, depending on the lowest main steam safety valve lift setpoint. In response to these issues, the PWROG performed additional analytical calculations using Westinghouse's seal leakage model (i.e., ITCHSEAL). These calculations included (1) benchmarking calculations against test data and (2) additional generic calculations for several groups of plants (categorized by similarity of first-stage seal leakoff line design) to determine the maximum leakage rates as well as the maximum pressures that may be experienced in the first-stage seal leakoff line piping.

During the audit review, Duke indicated that HNP is relying on the generic Westinghouse RCP seal leakage calculations that have been performed by the PWROG. The generic PWROG calculations audited by the NRC staff include proprietary reports PWROG-14015-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power," Revision 0 [Reference 29]; PWROG-14027-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power, Task 3: Evaluations of Revised Seal Flow Rate on Time to Enter Reflux Cooling and Time at which the Core Uncovers," Revision 3 [Reference 38]; and PWROG-14074-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power," Revision 0 [Reference 79].

These generic reports classify HNP in the first generic analysis category (i.e., Category I) specified in NSAL 14-1 [Reference 67]. As noted above, the generic analysis category definitions used in these reports were established based on the hydraulic characteristics of the first-stage seal leakoff line. Duke provided further information which confirmed that the leakoff line hydraulic characteristics for HNP are bounded by the assumed characteristics analyzed for Category I.

In support of beyond-design-basis mitigating strategy reviews, the NRC staff performed an audit of the PWROG's generic effort to determine the expected seal leakage rates for Westinghouse RCPs under loss-of-seal-cooling conditions. A key audit issue was the capability of Westinghouse's ITCHSEAL code to reproduce measured seal leakage rates under representative conditions. Considering known testing and operational events according to their applicability to the thermal-hydraulic conditions associated with the analyzed ELAP event, the benchmarking effort focused on comparisons of ITCHSEAL simulations to RCP seal leakage testing performed in the mid-1980s at Électricité de France's Montereau facility. Comparisons of analytical results to the Montereau data indicated that, while the ITCHSEAL code could not simultaneously obtain good agreement for both seal leakage and seal leakoff line pressure, the leakage rate simulated by ITCHSEAL could be tuned to reproduce the measured seal leakage rate data. Subsequent to the benchmarking effort, data from an additional RCP seal leakage test at the Montereau facility that had not been documented in WCAP-10541-P [Reference 66], was brought to the NRC staff's attention. The leakage rate during this test was significantly higher than that of the test in WCAP-10541-P [Reference 66] that had been used to benchmark the ITCHSEAL code. However, conservative margin was identified in the ITCHSEAL analyses (e.g., PWROG-14015-P, PWROG-14027-P), which the NRC staff expects should offset the potential for increased leakage rates observed in the additional Montereau test.

The NRC staff also audited information available from Westinghouse and AREVA associated with corrosion of the silicon nitride ceramic used to fabricate the first-stage seal faceplates currently used in Westinghouse-designed RCP seals. This specific material corrosion phenomenon was not present in the Montereau testing because that test article's faceplates were fabricated from aluminum oxide, consistent with the seals of actual Westinghouse-designed RCPs of that era. Hydrothermal corrosion of silicon nitride (likely assisted by flow erosion) became a focus area because test data from AREVA indicates that seal leakage could eventually exceed the values assumed in the analyses due to this phenomenon. Academic research reviewed by the industry and NRC staff associated with the general phenomenon indicates that the corrosion rate is temperature dependent. The PWROG is currently developing guidance for licensees to follow in order to reduce or eliminate seal face degradation and to respond to higher RCP seal leakage rates.

From the limited information available regarding the AREVA tests, as well as several sensitivity calculations performed during the audit, the NRC staff concluded that (1) the leakage rate for silicon-nitride RCP seals may be lower initially than had been predicted analytically by the PWROG's generic analysis using ITCHSEAL, (2) the RCP seal leakage rate during Phase 2 and Phase 3 of the ELAP event may increase beyond the long-term rate predicted analytically by the PWROG, and (3) certain aspects of the seal behavior observed in the AREVA tests did not appear consistent with the expected behavior based on models and theory that formed the basis for the WCAP and PWROG reports discussed above. However, because HNP's mitigating strategy plans for a more extensive RCS cooldown than is typical, the impact of the corrosion phenomenon is expected to be correspondingly reduced. In particular, according to currently available information, the NRC staff does not expect that silicon nitride faceplate degradation under ELAP conditions would lead to RCS leakage in excess of the makeup rate that HNP can provide using the FLEX equipment specified in its FIP.

The generic seal leakoff analysis discussed above assumes no failure of the seal design, including the elastomeric o-rings. During the audit review, Duke confirmed that all installed RCP seal o-rings at Harris are the high-temperature-qualified 7228-C type, and that only equivalent or better o-rings will be used in the future. Based on these factors, the NRC staff has confidence that o-ring failure would not be expected for HNP under analyzed ELAP conditions.

In order to ensure that the generic Category I (from NSAL-14-1) leakage rates are applicable to HNP, the NRC staff requested during the audit that Duke confirm that applicable portions of the first-stage seal leakoff line piping can withstand the maximum pressure that would be experienced during an ELAP event. According to generic calculations performed by Westinghouse using the ITCHSEAL code, Category I plants would be expected to experience choked flow at the flow-measurement orifice in the first-stage seal leakoff line, even after completion of the RCS cooldown. Therefore, to support application of the generic Category I leakage rates, it is necessary for Duke to demonstrate that a rupture in the pressure boundary of leakoff line piping and components upstream of the flow orifice would not occur at HNP. Duke has calculated that the first-stage seal leakoff line can withstand pressures of at least 2,500 psia (at an RCS cold leg temperature of 568°F), which exceeds the expected peak RCS pressure during an ELAP event and is equivalent to the RCS design pressure. Based upon this information, the NRC staff has confidence that the seal leakoff line would not rupture under analyzed ELAP conditions.

During the audit review, Duke further confirmed that, following the loss of seal cooling that results from the ELAP event, seal cooling will not be restored. The NRC staff considers this practice appropriate because it prevents thermal shock, which, as described in Information Notice 2005-14, "Fire Protection Findings on Loss of Seal Cooling to Westinghouse Reactor Coolant Pumps," could lead to unanalyzed increases in seal leakage.

In conjunction with the revised seal leakage analysis that Westinghouse performed, as described above, the PWROG's generic effort also sought to demonstrate that the second-stage seal will remain fully closed during the ELAP event. If the second-stage seal were to open, an additional term accounting for leakage past the second-stage seal and up the pump shaft could add to the first-stage seal leakoff line flow that has been considered in Duke's evaluation discussed in this section. Previous calculations documented in WCAP-10541-P [Reference 66] indicated that second-stage seal closure could be maintained under the set of station blackout conditions and associated assumptions analyzed therein. Recent calculations performed by Westinghouse and AREVA in support of PWR licensees' mitigating strategies indicated that both vendors also expected the second-stage seals essentially to remain closed throughout an ELAP event, even when the RCS is cooled down and depressurized in accordance with a typical strategy. Contrary to these analytical calculations, two recent RCP seal leakage tests performed by AREVA have indicated that the second-stage seals could open and remain open under ELAP conditions. While this behavior is not well understood, the leakage from the second stage seal did not appear to increase the total leakage rate from the RCP seals. Furthermore, during the audit review, Duke confirmed that HNP would cool the RCS down to 350°F within 24 hours, which satisfies recommendations from Westinghouse in Technical Bulletin 15-1 associated with maintaining long-term integrity of the second-stage seal.

The NRC staff further noted a number of conservatisms in the licensee's evaluation of RCP seal leakage, most significantly;

- The licensee's plan to initiate FLEX RCS injection at 10.5 hours provides a margin of over 4 hours to the calculated time of 14.8 hours when the RCS is considered to enter the reflux cooling mode (i.e., natural circulation is considered to be lost).
- The licensee's calculated time for entering reflux cooling conservatively neglected certain beneficial characteristics of HNP and its mitigating strategy relative to the reference plant, particularly HNP's more extensive RCS depressurization.

- The PWROG's generic ITCHSEAL calculations contain known conservatisms, such as the assumption of very low flow resistance in the generic leakoff line configuration analyzed to determine maximum seal leakage rates.

- Although entry into reflux cooling is undesirable and has not been fully analyzed in the context of the ELAP event, the NRC staff expects that the use of this threshold as an acceptance criterion for the time to establish RCS makeup provides significant margin to uncovering and severely damaging the core.

Based upon the NRC staff's understanding of RCP seal behavior, as reflected in the technical evaluation above, the NRC staff agrees with the licensee's conclusion that expected leakage from RCP seals would not result in HNP losing natural circulation during the analyzed ELAP event. Therefore, the NRC staff concludes that the licensee's strategy for ensuring adequate RCS inventory should meet the order's requirements to maintain or restore core cooling for the beyond-design-basis ELAP event.

3.2.3.4 Shutdown Margin Analyses

In an analyzed ELAP event, the loss of electrical power to control rod drive mechanisms is assumed to result in an immediate reactor trip with the full insertion of all control rods into the core. The insertion of the control rods provides sufficient negative reactivity to achieve subcriticality at post-trip conditions. However, as the ELAP event progresses, the shutdown margin for PWRs is typically affected by several primary factors:

- the cooldown of the RCS and fuel rods adds positive reactivity
- the concentration of xenon-135
    - initially increases above its equilibrium value following reactor trip, thereby adding negative reactivity
    - peaks at roughly 12 hours post-trip and subsequently decays away gradually, thereby adding positive reactivity
- the injection of borated makeup from passive accumulators due to the depressurization of the RCS, which adds negative reactivity

At some point following the cooldown of the RCS, PWR licensees' mitigating strategies generally require active injection of borated coolant via FLEX equipment. In many cases, boration would become necessary to offset the gradual positive reactivity addition associated with the decay of xenon-135, but in any event, borated makeup would eventually be required to offset ongoing RCS leakage. The necessary timing and volume of borated makeup depend on the particular magnitudes of the above factors for individual reactors.

The specific values for these and other factors that could influence the core reactivity balance that are assumed in the licensee's current calculations could be affected by future changes to the core design. However, NEI 12-06 [Reference 6], Section 11.8 states that "[e]xisting plant configuration control procedures will be modified to ensure that changes to the plant design ...

will not adversely impact the approved FLEX strategies." Inasmuch as changes to the core design are changes to the plant design, the NRC staff expects that any core design changes, such as a core reload analysis, will be evaluated to determine that they do not adversely impact the approved FLEX strategies, especially the analyses that demonstrate that recriticality will not occur during a FLEX RCS cooldown. The NRC staff discussed this issue with the licensee during the audit and further observed that the licensee's shutdown margin calculation includes a precaution and limitation addressing the calculation's applicability to future operating cycles.

The NRC staff requested that the industry provide additional information to justify that borated makeup would adequately mix with the RCS volume under natural circulation conditions potentially involving two-phase flow. In response, the Westinghouse PWROG submitted Westinghouse LTR-FSE-13-46, Revision 0, P-Attachment, "Westinghouse Response to NRC Generic Request for Additional Information (RAI) on Boron Mixing in Support of the Pressurized Water Reactor Owners Group (PWROG)," dated August 16, 2013 [Reference 90], (Enclosure Withheld from Public disclosure for Proprietary Reasons). This position paper provided test data regarding boric acid mixing under single-phase natural circulation conditions and outlined applicability conditions intended to ensure that boric acid addition and mixing during an ELAP would occur under conditions similar to those for which boric acid mixing data is available. By letter dated January 8, 2014 [Reference 46], the NRC staff endorsed the above position paper with three conditions:

- The required timing and quantity of borated makeup should consider conditions with no RCS leakage and with the highest applicable leakage rate.
- Adequate borated makeup should be provided either (1) prior to the RCS natural circulation flow decreasing below the flow rate corresponding to single-phase natural circulation, or (2) if provided later, then the negative reactivity from the injected boric acid should not be credited until one hour after the flow rate in the RCS has been restored and maintained above the flow rate corresponding to single-phase natural circulation.
- A delay period adequate to allow the injected boric acid solution to mix with the RCS inventory should be accounted for when determining the required timing for borated makeup. Provided that the flow in all loops is greater than or equal to the corresponding single-phase natural circulation flow rate, a mixing delay period of one hour is considered appropriate.

During the audit review, Duke confirmed that Harris will comply with the August 16, 2013, position paper [Reference 90] on boric acid mixing, including the conditions imposed in the NRC staff's corresponding endorsement letter, dated January 8, 2014 [Reference 46].

According to the FIP [Reference 18], injection of borated water from the BAT into the RCS will be initiated no later than 10.5 hours into the event. The licensee's analysis has determined that this will ensure makeup is initiated prior to a transition to reduced loop flow conditions. Calculation HNP-F/NFSA-0240 Revision 1, dated March 5, 2015 [Reference 57], shows that 4,800 gallons of 7,000 ppm borated water is sufficient to ensure long-term subcriticality at the lower bound RCS cooldown temperature of 350°F, and administrative controls ensure that this volume will remain valid for future core designs. By using the 40-gpm FLEX RCS pump, injection of the required boration volume will be completed no later than 19 hours into the event, even under conservative assumptions regarding accumulator injection and RCS venting. Completion of boration by 19 hours under these conservative assumptions provides significant margin with respect to the prevention of inadvertent re-criticality during the ELAP event. Upon completion of RCS boration from the BAT, the credited source for long term RCS makeup is the RWST, which has a credited available inventory of 187,000 gallons and a boron concentration of at least 2,400 ppm.

Therefore, based on the evaluation above, the NRC staff concludes that the planned operator actions should result in acceptable shutdown margin during the BDBEE.

3.2.3.5 FLEX Pumps and Water Supplies

Duke's FLEX strategy relies on three different portable pumps during Phase 2. A FLEX AFW Pump provides cooling water to the SGs taking suction from a number of water sources outlined in Section 3.10. Duke also relies on a FLEX RCS Pump to provide low flow, high pressure makeup to the RCS, and on a FLEX ESW Pump to provide long-term AFW and SFP makeup until higher quality water sources become available. The NRC staff noted that Duke identified the performance criteria (e.g., flow rate, discharge pressure, total dynamic head) for its FLEX Phase 2 portable pumps in Section 2.3.10 of its FIP [Reference 18]. The NRC staff noted that the performance criteria for the FLEX Phase 2 portable pumps are consistent with the FLEX Phase 3 portable pumps performance criteria.

The FLEX AFW Pump can take suction from the CST, RWST, RMWST or ESW and discharges to the AFW system via the discharge of either A or B motor-driven AFW pumps. Two cart-mounted, motor-driven centrifugal FLEX AFW pumps are stored in the Tank Building. These cart-mounted pumps are rolled into place and connected to the mechanical and electrical systems necessary for operation. A single FLEX AFW Pump has the capacity to provide 325 gpm at 363 psig to all three SGs. A single pump provides full capability to feed all three generators so Duke has two portable FLEX AFW Pumps to satisfy the N+1 requirement, outlined in NEI 12-06 [Reference 6]. During the audit review, Duke provided the hydraulic calculation HNP-M/FLEX-0006, "FLEX AFW Pump Sizing and Pressure Drop Calculation," Revision 1, [Reference 47], which determined the necessary pump performance criteria for the portable FLEX AFW Pump to support the licensee's core cooling strategies. The NRC staff noted that Duke's calculation relied upon the hydraulic modeling program PROTO-FLO to create a hydraulic model using FLEX strategies and that the piping models included the installed AFW system piping and the portable hoses, valves and adaptors required to connect the FLEX AFW Pump to the AFW system. This calculation assessed different possible lineups based on different suction sources, and the different connection points.

The FLEX RCS Pump can take suction from the BAT, ASI tank or RWST and can discharge to different connections in the charging system, which will inject into the RCS, using independent SI piping trains. Two cart-mounted, motor-driven positive displacement pumps are stored in the Tank Building and can provide 40 gpm at 1600 psig. A single pump provides full RCS injection capability so Duke has two portable FLEX RCS Pumps to satisfy the N+1 requirement, outlined in NEI 12-06 [Reference 6]. During the audit review, the licensee provided Engineering Change

(EC) 91680, "FLEX RCS Injection Evaluation," Revision 5, [Reference 48] which determined the pump performance criteria taking into account the different flow paths and suction sources.

Lastly, the licensee relies on a FLEX ESW Pump to provide long-term AFW and SFP makeup. The licensee procured two trailer-mounted, diesel-driven centrifugal pumps to provide ESW flow following an ELAP. During the audit review, the licensee provided Calculation HNP-M-FLEX-0008, "ESW Pump and Hose Sizing for FLEX Strategies," Revision 0 [Reference 63] that describes the flow qualification required for the FLEX ESW pump. The model takes into account both the primary suction using the ESW intake structure and putting the suction hose directly in the Auxiliary Reservoir. The pump can provide up to 3000 gpm at 150 psig, which the calculation showed was sufficient. This pump provides an NPSH margin of 8 ft. from normal level in the Auxiliary reservoir. One pump will provide the necessary flow so having two fulfills the N+1 requirement specified in NEI 12-06 [Reference 6].

The NRC staff walked down the FLEX AFW and RCS deployment paths. The FLEX RCS and AFW pumps are stored on the 236' level of the Tank Building (Seismic Category 1) on a cart, and will be rolled approximately 200 ft. into place by operators and placed in the vicinity of the AFW pump and the charging pumps in the RAB. From those points, multiple connections can be made from the pump locations to multiple trains of AFW and Charging. The NRC staff also walked down the placement of the FLEX ESW Pump at the intake structure. This pump can be placed outside of the ESW intake building where the suction and the discharge will be routed into the ESW building. The ESW FLEX pump can also take suction directly from the Auxiliary Reservoir if the ESW intake structure is blocked or otherwise disabled. The NRC staff walked down the hose deployment routes and found them to be consistent with the hydraulic analyses discussed above

Based on its review, the NRC staff concludes, if implementation is performed as described, that the licensee has demonstrated that its FLEX portable pumps are capable of supporting the water make-up to the SGs, RCS and refilling the CST.

3.2.3.6 Electrical Analyses

Duke's electrical FLEX strategies are identical for maintaining or restoring core cooling, containment, and SFP cooling, except as noted in Sections 3.3.4.4 and 3.4.4.4 of this SE. Furthermore, the electrical coping strategies are the same for all modes of operation.

According to the FIP, the Phase 1 electrical strategy involves completing a shallow station blackout (SBO) load shed within 1 hour and a deep load shed within 2 hours of the ELAP. During the first phase of the ELAP event, HNP will be relying on the Class 1E station batteries to cope until additional power supplies (i.e., FLEX DGs) can be aligned and connected to the HNP electrical distribution system (Phase 2).

During the audit, NRC staff reviewed Calculation Number E4-0008, "125VDC 1E Battery Sizing and Battery/Panel Voltages for Station Blackout," Revision 7, [Reference 68] which verified the capability of the dc system to supply the required loads during the first phase of the HNP FLEX mitigation strategy plan for an ELAP as a result of a BDBEE. The HNP analysis identified the required loads and their associated ratings (amperage and minimum voltage) and loads that would be shed to ensure battery operation for at least 10 hours. Duke expects that power will be restored to the battery charger within 8 hours. Duke endorsed NEI white paper, "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," dated August 27, 2013 [Reference 70], which was endorsed by the NRC staff on September 16, 2013 [Reference 71].

In addition to the NEI 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 purpose of this testing was to examine whether existing vented lead acid batteries can function beyond their defined design-basis (or beyond-design-basis if existing SBO coping analyses were utilized) duty cycles in order to support core cooling. The study evaluated battery performance availability and capability to supply the necessary dc loads to support core cooling and instrumentation requirements for extended periods of time.

The testing provided an indication of the amount of time available (depending on the actual load profile) for batteries to continue to supply power beyond the original duty cycles for a representative plant. The testing also demonstrated that battery availability can be significantly extended using load shedding techniques to allow more time to recover ac power. The testing further demonstrated that battery performance is consistent with manufacturer performance data. According to the NUREG, the projected availability of a battery can be accurately calculated using the Institute of Electrical and Electronics Engineers (IEEE) Standard 485-2010, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," or using an empirical algorithm described in the report.

Based on the information contained in NUREG/CR-7188, and its review of the licensee's analysis, the battery vendor's capacity and discharge rates for the batteries, and procedures EOP-ECA-0.0, "Loss Of All AC Power," Revision 3 [Reference 40] and FSG-004, "ELAP DC Bus Load Shed/Management," Revision 0 [Reference 69], the NRC staff found that Duke's load shed strategy is acceptable and that the batteries should have sufficient capacity to supply power to required loads for at least 10 hours.

The licensee's transition from Phase 1 to Phase 2 (repowering station battery chargers) includes aligning and placing into service one (of two) 480 Volt (V) ac 3-phase 830 kilowatt (kW) pre-staged FLEX DGs located in the EDG Building Bay 2B. The second permanently pre-staged FLEX DG provides backup capability. One 480 Vac FLEX DG would provide power to vital battery chargers, 480V switchgear (SWGR) 1A3-SA or 1B3-SB, the RAB FLEX MCC, and the necessary EDG Building house loads through the Diesel Generator Building MCC, FLEX RCS Pump, FLEX AFW Pump, SG PORV hydraulic pump motors, portable fans and heaters, and alternate power to MCC buckets for valve operation. The licensee's strategy assumes that the FLEX DGs will be deployed and will repower dc buses and the Battery Chargers within 8 hours (i.e., before the station batteries fully discharge).

The NRC staff reviewed Calculation Number E-6009, "Fukushima Diesel Generator Sizing Calculation," Revision 1 [Reference 91], the FLEX DG manufacturer specification sheets, conceptual single line electrical diagrams, the separation and isolation of the FLEX DGs from the Class 1E EDGs, and procedures that direct operators how to align, connect, and protect associated systems and components. Based on its review of the calculations, the NRC staff confirmed that the FLEX DGs should have sufficient capacity and capability to supply the necessary loads following a BDBEE.

For Phase 3, Duke plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources. The offsite resources that will be provided by the NSRCs includes a 1-megawatt (MW) 480 Vac turbine generator. The NSRC FLEX DG can be connected via the 480 Vac patch panels located in the EDG Building Bay 2B. The connection points on the patch panels are compatible with the NSRC FLEX DG cables. Based on its review, the NRC staff finds that the 480 Vac FLEX DG being supplied from the NSRCs should provide adequate power to enable HNP to maintain or restore core cooling, SFP cooling, and containment indefinitely following a BDBEE.

3.2.4 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately, should maintain or restore core cooling following a BDBEE consistent with NEI 12-06, 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 [Reference 6], Table 3-2 and Appendix D summarize one acceptable approach consisting of three strategies 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 SFP cooling piping or other alternate location capable of exceeding the boil-off rate for the design-basis heat load; and alternatively 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 to account for overspray). This approach will also provide a vent pathway for steam and condensate from the SFP.

3.3.1 Phase 1

The FIP, dated July 10, 2015 [Reference 18], states that no action is required for phase 1 SFP makeup because the time to uncover the fuel is sufficient to enable deployment of Phase 2 equipment. The FIP also states that the time to reach 10 feet above the top of the fuel is 35.9 hours. Additionally, Duke can gravity drain from the RWST to the SFP, using installed plant piping, if the refueling floor is not accessible, and this strategy does not rely on any portable equipment.

3.3.2 Phase 2

Duke's Phase 2 SFP cooling strategy consists of pressurizing the ESW header with FLEX ESW Pump. The ESW can be aligned to the safe-shutdown earthquake (SSE) Fire Protection header. Hoses would be routed from the SSE header hose stations to the SFP refueling floor and into the pool. Additionally, Duke can provide up to 250 gpm of spray flow to the SFP using their Emergency Diesel Makeup Pump (EDMP).

3.3.3 Phase 3

Duke will continue to use its Phase 2 strategy into Phase 3 with the NSRC pump replacing the onsite FLEX ESW Pump if necessary. Duke will receive the water treatment equipment at HNP from the NSRC to ensure a cleaner long-term water source.

3.3.4 Staff Evaluations

3.3.4.1 Availability of SSCs

3.3.4.1.1 Plant SSCs

Condition 6 of NEI 12-06 [Reference 6], 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.

Harris has four different SFPs in the Fuel Handling Building (FHB), which are Pool A, B, C, and D. As described in the licensee's FIP, the licensee's Phase 1 SFP cooling strategy does not require any anticipated actions. However, the licensee does establish a ventilation path to cope with temperature, humidity, and condensation from evaporation and/or boiling of the SFP. FSG-005, "Initial Assessment and FLEX Equipment Staging," Revision 0 [Reference 42] directs the operators to establish a FHB vent pathway as discussed in FSG-011, "Alternate SFP Makeup and Cooling," Revision 0 [Reference 41]. Guideline FSG-011 contains precautions to establish a vent path within 2-3 hours following the ELAP. The operators are directed to open the rail bay door on the 261' for an inlet and two personnel doors on the refueling floor (286') as an exhaust.

Additionally, as stated above, Duke will not need to take any action to cool the SFP until 39 hours into the event; however, Duke can gravity feed from the RWST to the SFP, using only installed equipment. As described in the Harris UFSAR Table 3.2.1-1, the RWST is a seismic Category I tank located in the Tank Building, which is a seismic Category I building and as described in the FIP [Reference 18], has a credited volume of 407,000 gallons for all events except a tornado missile event in which case approximately 187,000 gallons is available. The Tank Building, RAB and the FHB are all Category I seismic, temperature controlled and tornado wind and missile protected. In EC 91701, "Seismically Upgrade SFP Piping for NTTF 4.2 (FLEX)," Revision 0 [Reference 43], Duke re-analyzed the only non-safety-related portion of the flow path from the RWST to the SFP, and upgraded this flow path to seismically robust.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves use of the FLEX ESW pump (or NSRC supplied pump for Phase 3) to pressurize either A or B ESW header. The ESW system can be cross-connected with the SSE fire protection header. Two SSE fire protection header hose stations are available on the 286' (refueling floor level) and two on the 261' level. A three inch hose is routed from any one of the four hose stations onto the refueling floor and then into the pool. As described in the Harris UFSAR Section 9.5.1, the post SSE fire protection standpipe and hose systems in portions of the RAB and FHB were analyzed for SSE loading and seismically supported to assure system pressure integrity. These hose stations are protected from all applicable hazards because they are located in the RAB and FHB, which are temperature controlled, seismic Category I, and tornado wind and missile protected. The NRC staff's evaluation of the robustness and availability of FLEX connections points for the FLEX ESW pump is discussed in SE Section 3.7.3.1. Furthermore, the NRC staff's evaluation of the robustness and availability of Auxiliary Reservoir for an ELAP event is discussed in SE Section 3.10.3.

3.3.4.1.2 Plant Instrumentation

In its FIP, the licensee stated that the instrumentation for SFP level will align with 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 480 V FLEX DGs. The NRC staff's review of the SFP level instrumentation, including the primary and back-up channels, the display to monitor the SFP water level and environmental qualifications to operate reliably for an extended period are discussed in Section 4 of this SE.

3.3.4.2 Thermal-Hydraulic Analyses

Duke keeps fuel in three separate SFPs, A, B and C. As described in CSD-EG-HNP-8888, the most limiting configuration occurs during a full core offload when A and B are connected and C is isolated. In its FIP, the licensee stated that during the most limiting conditions and configuration, the time when boil-off decreases the water level to 10 feet above the top of the SFP racks is 35.9 hours and the time to reach the top of active fuel is 61.9 hours. According to CSD-EG-HNP-8888, the time to boil following an emergency core offload to pool A and B is 9.7 hours and will require a makeup flow of 90 gpm resulting from a maximum heat load of 46.23 MBTU (Million British Thermal Unit)/hour. Pool C does not receive fuel during a core offload so the time to boil for pool C remains 60.4 hrs, and 15 gpm of makeup flow to pool C is needed to make up for boiling losses. The total required makeup rate during the most limiting conditions stated in the FIP is 105 gpm.

3.3.4.3 FLEX Pumps and Water Supplies

As described in the FIP, Duke's SFP cooling strategy relies on one of two FLEX ESW pumps to provide SFP makeup during Phase 2. In the FIP, Section 2.4 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX ESW pump. In the Harris FIP, the licensee stated the FLEX ESW Pump could provide up 260 gpm makeup to the SFP via any one of the four SSE fire protection hose stations. However, the NRC staff noted that these FLEX ESW pumps could not provide SFP spray flow. The NRC staff noted that the performance criteria for the FLEX Phase 3 portable pumps are consistent with the FLEX Phase 2 portable pump capacities. The licensee can also add water to the SFP by gravity drain from the RWST. Harris Document CSD-EG-HNP-8888 states that RWST can gravity drain to the SFP providing up to 3700 gpm at initiation of the system alignment. This flow will decrease proportionately as RWST level decreases until RWST level reaches approximately 40 percent.

As described in a supplement to the Harris EA-12-049 Compliance letter, dated June, 29, 2016 [Reference 94], the licensee relies on a separate EDMP to provide SFP spray flow. The EDMP is a trailer mounted, diesel driven pump capable of providing greater than 300 gpm to the SFP via spray nozzles,which exceeds the required 250 gpm performance attribute specified by NEI 12-06 [Reference 6] Table D-3. This pump is protected against external hazards and stored in the EDG Building 2A Bay. However, the licensee only has one (N) EDMP and thus is an alternative to NEI 12-06 [Reference 6] which specifies N+1 capability. See Section 3.14 for further discussion of this alternative.

Additionally, during the onsite audit, the NRC staff performed a walkdown of the licensee's SFP cooling FLEX strategies and noted that the point of deployment for the portable FLEX pumps, hose routing and deployment connection points (primary and alternate) were consistent with the licensee's hydraulic analysis. Procedure FSG-011, "Alternate SFP Makeup and Cooling," Revision 0, [Reference 41] provides procedural guidance for SFP cooling. The procedure contains precautions to route hoses early to minimize personnel exposure to heat, humidity and radiation. The procedure also states that normally all SFPs at Harris are connected (normal configuration) and provides guidance for conditions where SFPs are isolated. This FSG also provides guidance for deployment of SFP makeup via hose monitor spray nozzles.

3.3.4.4 Electrical Analyses

The NRC staff's review of the licensee's electrical strategies, which includes the SFP cooling strategy, is discussed in detail in Section 3.2.3.6.

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 a BDBEE consistent with NEI 12-06, 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 [Reference 6], Table 3-2 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.

In accordance with NEI 12-06 [Reference 6], Duke performed a containment evaluation, EC 91686, "Containment Temperature and Pressure Evaluation for NTTF 4.2 FLEX," Revision 1 [Reference 74], which was based on the boundary conditions described in Section 2 of NEI 12-06 [Reference 6]. Calculation, HNP-M/FLEX-0009, "FLEX ELAP Containment Temperature Pressure Analysis", Revision 0 [Reference 75], supporting this evaluation, concluded that the containment parameters of pressure and temperature remain well below the respective design limits of 45 psig (UFSAR Section 6.2, Table 6.2.1-3, Principle Containment Design Parameters) and 380°F (Technical Specification section 5.2.2) for more than 7 days. Based on 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.

The NRC staff reviewed evaluation EC 91686 and Calculation HNP-M/FLEX-0009. This calculation used the GOTHIC computer code (Generation of Thermal-Hydraulic Information for Containments). The calculation forecasted the Containment pressure reaches 15.3 psig 12-hours after the onset of the ELAP event. The pressure drops to 9 psig after 2 days and increases to 13.4 psig at the end of 10 days remaining well below the design-basis limit of 45 psig. The calculation also indicates the temperature will reach roughly 188°F at the end of 10 days, below the design limit of 380°F (Technical Specification Section 5.2.2).

3.4.1 Phase 1

Duke's containment analysis shows that there are no Phase 1 actions required. Procedure EOP-ECA-0.0, "Loss of all ac Power" [Reference 40] verifies containment isolation.

3.4.2 Phase 2

Duke's containment analysis shows that there are no Phase 2 actions required. Conservatively, Duke's FLEX strategy for containment integrity at HNP will provide the capability for containment cooling within 48 hours following an ELAP. Guideline FSG-012, "FLEX Containment Cooling", Revision 0 [Reference 76], provides guidance for implementing a long-term strategy for removing heat from containment as required. Long-term Containment cooling can be established by repowering one train of ESW along with a train of Containment Fan Coolers. Guideline FSG-012 cautions that if a SFP Cooling Pump is running on FLEX power, that pump must be stopped before starting a containment cooling fan to prevent overloading the FLEX Electrical Distribution. An alternate method for Containment cooling is to spray the exterior of the Containment shell with water using Fire Department Fire Tankers along with elevated oscillating monitors. Guideline FSG-012 also verifies the Containment is isolated from compressed air sources.

3.4.3 Phase 3

Guideline FSG-012 provides guidance for implementing a long-term strategy for removing heat from the containment, as required. Long-term Containment cooling can be established by repowering one train of ESW along with a train of Containment Fan Coolers. An alternate method for Containment cooling, is to spray the exterior of the Containment shell with water using Fire Department Fire Tankers along with elevated oscillating monitors. Guideline FSG-012 can be implemented using either FLEX equipment or equipment from the NSRC.

3.4.4 Staff Evaluations

3.4.4.1 Availability of SSCs

Guidance document NEI 12-06 [Reference 6] 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

Duke uses a large, dry concrete containment building, with its associated Containment Isolation System. The Concrete Containment Structure is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel.

Long-term containment cooling can be established by repowering one train of ESW along with a train of Containment Cooling System (CCS). Harris UFSAR Section 6.2.2 (amendment 56) indicates each train of CCS includes two safety-related fan cooler units. Safety-related portions of the CCS are designed to the requirements of Safety Class 2, Seismic Category I. The ESW is designed (UFSAR 9.2.5-3, amendment 56) to withstand, or be protected from the effects of, a SSE, a design-basis tornado, maximum flood levels, and high energy line break(s) without loss of safety function. The ESW for the CCS is established in accordance with FSG-005, "Initial Assessment and FLEX Equipment Staging", Revision 0 [Reference 42].

These are engineered safety systems and meet the qualification requirements for seismic Category I; therefore, in accordance with NEI 12-06 [Reference 6], it is considered to be available equipment following an ELAP event.

3.4.4.1.2 Plant Instrumentation

Consistent with Table 3-2 of NEI 12-06 [Reference 6], Revision 0, Duke stated, in its FIP, the key parameter for the containment integrity function is containment pressure, which can be obtained from the following instrumentation:

- PT-950 powered by instrument bus IDP-1A-SI
- PT-951 powered by instrument bus IDP-1B-SII
- PT-952 powered by instrument bus IDP-1A-SIII
- PT-953 powered by instrument bus IDP-1B-SIV

These parameters are expected to be available by connection to the safety train Instrumentation Buses. The Instrumentation buses IDP-1A-SI and IDP-1A-SIII are normally energized by safety-related 7.5kVA Channel I & III UPS inverters with dc input from the 1E 125VDC bus DP-1A-SA and battery 1A-SA, while IDP-1B-SII and IDP-1B-SIV are normally energized by safety-related 7.5kVA Channel II & IV UPS inverters with dc input from the 1E 125VDC bus DP-1B-SB and battery 1B-SB. Duke reviewed the environmental qualification for containment instrumentation and concluded that all instrumentation is qualified for the potential conditions from the postulated event.

If vital instrumentation and controls are lost, the alternate strategy is to obtain key parameter readings using portable instruments.

Containment temperature is not normally available during an ELAP. However, containment temperature may be obtained using temperature indicators TI-7541 and TI-7542. These temperature indications are powered from circuits de-energized in accordance with FSG-004, "ELAP DC Bus Load Shed/Management," Revision 0 [Reference 69]. One of these circuits would need to be re-energized to obtain containment temperature.

3.4.4.2 Thermal-Hydraulic Analyses

The NRC staff reviewed Calculation HNP-M/FLEX-0009, "FLEX ELAP Containment Temperature Pressure Analysis", Revision 0 [Reference 75]. This calculation used the GOTHIC Computer Code, Version 8.0. Two cases were run using different rates of RCS leakage. For both cases, initial containment pressure was assumed to be 1.6 psig and initial containment temperature was assumed to be 120°F. For both cases the calculation modeled the containment pressure and temperature over a 10-day period during an ELAP event. The event timeline was developed based on WCAP-17601, "Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering, and Babcock & Wilcox", Revision 1 [Reference 96].

Case 1 assumed 12.5 gpm of RCS leakage per pump through the RCP seals until the RCS is cooled down to 350°F, at which time the leakage was reduced to 1.9 gpm. The RCS unidentified leakage was assumed to be 1 gpm.

For Case 1, the containment pressure was calculated to be 9.9 psig 12 hours after the onset of the ELAP event. The containment pressure decreases to 6.1 psig by 2 days and increases to 7.8 psig by the end of the 10 days. The containment temperature reaches 182°F by 12 hours. After this time, temperature falls to 150°F by 2 days and slowly increases to 162° by the end of 10 days.

Case 2 is the same as Case 1 with 21 gpm of RCS leakage per pump RCP seals until the RCS is cooled down to 350°F. The RCS unidentified leakage was assumed to be 1 gpm. The licensee considered this case conservative based on industry document PWROG 14015-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power Task 2: Determine Seal Flow Rates", Revision 0.

For Case 2, the containment pressure was calculated to be 15.3 psig 12 hours after ELAP event. The pressure decreases to 9.0 psig by 2 days and increases to 13.4 psig by the end of the 10 days. The containment temperature reaches 206°F by 12 hours. After this time, temperature falls to 169°F by 2 days and slowly increases to 188° by the end of 10 days.

The temperature and pressure behavior is a result of some modeling assumptions. The RCS make-up was assumed to be initiated at 12 hours. Prior to this, the RCP seals were treated as uncovered, such that the seal leakage consists as steam. After 12 hours, the seals were treated as covered, such that the seal leakage consists of water that partially flashes to steam once it is exposed to containment pressure.

The Containment building remains below the internal design limits of 45 psig and 380°F for either case.

3.4.4.3 FLEX Pumps and Water Supplies

One FLEX pump or one NSRC pump will be used to re-pressurize the ESW. ESW is used as the cooling medium for the containment fan coolers. Since cooling is not required for some time after the onset of the event (approximately after 10 days) the Technical Support Center staff will be able to determine what equipment to use based on plant conditions and available equipment, including and plant equipment returned to service. Containment cooling would run for a short period of time to reduce the Containment temperature and pressure.

3.4.4.4 Electrical Analyses

According to its FIP, Duke performed a GOTHIC analysis to determine the temperature and pressure increase resulting from an ELAP during a BDBEE. Based on the results of this analysis, required actions to ensure maintenance of containment integrity and required instrumentation function have been developed. Duke concluded that no actions were required as part of Phase 1 or Phase 2, aside from monitoring containment pressure. However, Duke's FLEX strategy involves starting a containment fan cooler within 48 hours of the ELAP. For Phase 3, containment cooling is maintained following the Phase 2 strategy. The licensee confirmed that the FLEX DGs have the necessary capacity to support the necessary equipment during Phase 2 and 3. One FLEX generator or one NSRC turbine generator can be used to power one train of containment fan coolers as needed for long-term containment heat removal.

3.4.5 Conclusions

Based on this evaluation, the NRC staff concludes that Duke has developed guidance that, if implemented appropriately, should maintain or restore containment functions following a BDBEE consistent with NEI 12-06, 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 [Reference 6] provide the methodology to identify and characterize the applicable BDBEE for each site. In addition, NEI 12-06 [Reference 6] provides a process to identify potential complicating factors for the protection and deployment of equipment needed for mitigation of site-specific BDBEE leading to an ELAP and the LUHS. 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 the guidance described in NEI 12-06 [Reference 6] 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.

References to external hazards within the licensee's mitigating strategies and this SE are consistent with the guidance in NEI-12-06 and the related interim staff guidance in JLD-ISG-2012-01 [Reference 7]. Coincident with the issuance of the order, on March 12, 2012 the NRC staff issued a request for information pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) [Reference 19] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in responses to the 50.54(f) letter, the requirements for Order EA-12-049 and related rulemaking to address BDBEE (see COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards," dated November 21, 2014). On March 30, 2015, the Commission provided guidance in the SRM to COMSECY-14-0037 [Reference 20]. The Commission approved the staff's recommendations that licensees need to address the reevaluated flooding hazards within their mitigating strategies for BDBEE, and that licensees may need to address some specific flooding scenarios that could significantly damage the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition.

The licensee submitted its flood hazard reevaluation report (FHRR) on March 12, 2013 [Reference 21], and the NRC staff completed its review of the report as documented by letter dated April 29, 2015 [Reference 36]. The licensee noted that the increased levels did not exceed the flood protection capabilities or did not impact safety-related equipment. Consequently, the licensee is required to submit an Integrated Assessment. Therefore, this SE makes a determination based on the OIP and FIP, and only notes the possibility of future actions by the licensee if the licensee's Integrated Assessment identifies mitigating actions that have to be implemented to address flooding.

In accordance with the 50.54(f) letter, licensees were also asked to provide a seismic hazard screening and evaluation report to reevaluate the seismic hazard at their site. The licensee submitted its seismic hazard and screening report (SHSR) on March 27, 2014 [Reference 22], and the NRC staff completed its review of the report, as documented by letter dated December 18, 2015 [Reference 92], and the results are discussed in Section 3.5.1 below. Therefore, this SE makes a determination based on the OIP and FIP, and only notes the possibility of future actions by the licensee based on the NRC's review of the licensee's high-frequency evaluation.

The characterization of the specific external hazards for the plant site is discussed below. In addition, Sections 3.5.1 and 3.5.2 summarizes the licensee's activities to address the 50.54(f) seismic and flooding reevaluations.

3.5.1 Seismic

In its FIP, Duke stated that the UFSAR SSE has a ground acceleration design value of 0.15g, and that seismic hazards are applicable to the HNP site. The current NRC terminology for the desig-basis earthquake (DBE) is the SSE.

As previously discussed, the NRC issued a 50.54(f) letter that required facilities to reevaluate the site's seismic hazard (i.e., NTTF Recommendation 2.1). In addition, the 50.54(f) letter requested that licensees submit, along with the hazard evaluation, an interim evaluation and actions planned or taken to address the reevaluated hazard where it exceeds the current design-basis.

Based on the results of the SHSR, submitted on March 27, 2014 [Reference 22], Duke screened-in only for a High Frequency Confirmation and screens out for the Expedited Seismic Evaluation Process, SFP, and a seismic risk evaluation. Duke stated in its SHSR, that the ground motion response spectrum (GMRS) only exceeds the SSE for high frequencies. This motion is considered to be non-damaging to components and structures that have strain or stress based potential failures modes.

By letter dated December 18, 2015 [Reference 92], the NRC staff completed its review of Duke's SHSR. The NRC staff concluded that the licensee conducted the hazard reevaluation using present-day methodologies and regulatory guidance. In addition, the licensee appropriately characterized the site given the information available, and met the intent of the guidance for determining the reevaluated seismic hazard. The NRC staff also concluded that the licensee provided an acceptable response reevaluated seismic hazard for HNP is suitable for other activities associated with the NTTF Recommendation 2.1, "Seismic."

By letter dated February 18, 2016 [Reference 97], the NRC staff confirmed that HNP met the 3.1.2 "limited High Frequency Exceedance Screening" critieria desribed in EPRI report 3002004396, thus is acceptable. In its letter, the NRC staff letter closed out it efforts associated with Phase 1 and Phase 2 of NTTF 2.1 "Seismic", thus no further response or regulatory acition are required. Duke has appropriately screened in this external hazard and identified the hazard levels for reasonable protection of the FLEX equipment.

3.5.2 Flooding

In its FIP, Duke stated that the external flooding hazard is not applicable for HNP. HNP is a dry site with a nominal plant elevation of 260 feet mean sea level (MSL) and a maximum water level, due to the probable maximum flood event, of 257.7 feet MSL.

On March 12, 2013, Duke submitted its FHRR [Reference 21] which showed that some of the flood-causing mechanisms are not bounded by the current plant-specific design-basis hazard. Duke stated that the increased levels did not exceed the flood protection capabilities or did not impact safety-related equipment; therefore no interim actions were planned.

Based on the results of its FHRR and in accordance with NRC staff issued guidance on the trigger conditions for performing an Integrated Assessment [Reference 32], Duke was required to perform an Integrated Assessment. Duke committed to performing an Integrated Assessment and submittal of a final report to the NRC before March 12, 2015. However, subsequent to the issuance of the 50.54(f) letter, the NRC issued a letter on May 26, 2015 [Reference 33], deffering, until further notice, the date for submitting the Integrated Assessment Reports.

Duke will complete the integrated assessment as requested in the NRC's 50.54(f) letter by performing either a focused evaluation or an integrated assessment documenting hazard response at the site for flood causing mechanisms that are not bounded by the current design basis, as described in the interim hazard letter. Guidance describing suitable methods to complete a focused evaluation or integrated assessment was submitted to the NRC by NEI in NEI 16-05, "External Flooding Integrated Assessment Guidelines." The licensee is expected to complete the focused evaluation by June 2017, or an integrated assessment by December 2018 if necessary, consistent with the guidance described in NRC's COMSECY-15-0019.

The NRC staff issued an RAI to the licensee on February 10, 2014 [Reference 34], requesting additional information regarding the FHRR. As a result of the request, Duke issued Revision 1 to its FHRR on April 1, 2015 [Reference 35], to address some administrative errors in the original submittal. The NRC staff completed its assessment of the FHRR on April 29, 2015 [Reference 36]. In its assessment, the NRC staff confirmed Duke's conclusions that the reevaluated flood hazard results for local intense precipitation, flooding from streams and rivers, and storm surge flooding mechanisms were not bounded by the current design-basis flood hazard; therefore, a focused evaluation or Integrated Assessment, including local intense precipitation, streams and rivers, and storm surge floods was expected to be submitted by the licensee; and the reevaluated flood-causing mechanism information was appropriate input to the focused evaluation or Integrated Assessment as described in NEI 16-05, as endorsed by the NRC in JLD-ISG-2016-01 [Reference 37]. Therefore, this SE makes a determination based on the OIP and FIP, and only notes the possibility of future actions by the licensee if the licensee's focused evaluation or Integrated Assessment identifies mitigating actions that have to be implemented to address flooding.

As the licensee's flooding reevaluation activities are completed, the licensee will enter appropriate issues into the corrective action program. The licensee has appropriately screened in this external hazard and identified the hazard levels for reasonable protection of the FLEX equipment.

3.5.3 High Winds

In NEI 12-06 [Reference 6], Section 7, it provides the NRC-endorsed screening process for evaluation of high wind hazards. This screening process considers the hazard due to hurricanes and tornadoes. The first part of the evaluation of high wind challenges is determining whether the site is potentially susceptible to different high wind conditions to allow characterization of the applicable high wind hazard. The second part is the characterization of the applicable high wind threat.

The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06 [Reference 6], Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009); if the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 miles per hour (mph) exceeds 10^{-6} per year probability, the site should address hazards due to extreme high winds associated with hurricanes.

The screening for high wind hazard associated with tornadoes should be accomplished by comparing the site location to NEI 12-06 [Reference 6], 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 10^{-6} per year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes.

In its FIP, the licensee stated that the high wind hazard is applicable for HNP. Harris UFSAR Section 2.1.1 indicates that HNP is located at latitude 35° 38' 0" N, and longitude 78° 57' 22" W. Duke added that according to NEI 12-06 [Reference 6], Figures 7-1 and 7-2, the location of HNP has a peak-gust wind speed of 160 mph and a tornado design wind speed of 200 mph. These values indicate that HNP has the potential to experience severe winds from hurricanes and tornadoes with the capacity to do significant damage, which are generally considered to be winds above 130 mph, as defined in NEI 12-06, Section 7.2.1.

The NRC staff compared the documented location for HNP with NEI, Figure 7-1 and verified that the site is in an area that has a frequency of recurrence of hurricanes with wind speeds in excess of 130 mph with less than 10^{-6} per year probability, which would screen in the high wind hazards, which were considered by Duke in developing the mitigation strategies.

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 [Reference 6], Section 8.2.1, all sites should consider the temperature ranges and weather conditions for their site in storing and deploying their 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, Duke stated that extreme cold (including snow and ice) hazard is applicable for HNP. Duke added that the location of HNP is latitude 35° 38' 0" N and longitude 78° 57' 22" W, and as shown in NEI 12-06 [Reference 6], Figure 8-1, this location is subject to significant snowfall accumulation and extreme low temperatures. Therefore, Duke must provide the capability to address the impedances caused by extreme snowfall. Harris is also in a region with Level 4 ice storm severity as depicted in NEI 12-06 [Reference 6], Figure 8-2, which is characterized as severe damage to power lines and/or the existence of large amounts of ice.

In its FIP, Duke indicated that the winter temperatures used for calculation of heat transmission across building exterior walls range from -2°F to 16°F (UFSAR Table 9.4.0-1). The lowest recorded temperature in the HNP area is -9°F, and this was the minimum temperature applied in ELAP extreme cold analysis and calculations used in development of the FLEX strategies. Harris UFSAR Section 2.4.7 states that the minimum average temperatures recorded at Raleigh Airport for the months of December, January, and February are 30.5°F, 30.0°F, and 31.1°F, respectively.

In summary, based on the available local data and Figures 8-1 and 8-2 of NEI 12-06 [Reference 6], 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 NEI 12-06 [Reference 6], Section 9 states that all sites will address high temperatures. Virtually every state in the lower 48 contiguous United States has experienced temperatures in excess of 110 °F. Many states have experienced temperatures in excess of 120 °F. In this case, sites should consider the impacts of these conditions on deployment of the FLEX equipment.

In its FIP, the licensee stated that the extreme heat hazard is applicable for HNP. Harris is located at latitude 35° 38' 0" N and longitude 78° 57' 22" W (UFSAR, Section 2.1.1.1). The summer temperatures used for calculation of heat transmission across building exterior walls range from 95°F to 105°F (UFSAR Table 9.4.0-1). In NEI 12-06 [Reference 6], Section 9.2 states that virtually every state in the lower 48 contiguous United States has experienced temperatures in excess of 110°F and many in excess of 120°F. In accordance with NEI 12-06 [Reference 6], Section 9.2, all sites will address high temperatures. Duke stated it used 110°F as an input to ELAP extreme heat analysis and calculations used in development of FLEX strategies.

In summary, based on the available local data and the guidance in Section 9 of NEI 12-06 [Reference 6], 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, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

The licensee stated in its FIP [Reference 18], as supplemented by letter dated June 29, 2016 [Reference 94], that FLEX equipment would be stored in the following buildings:

a. EDG Building – Bays 2A and 2B provide protected storage for FLEX equipment against all applicable extreme external hazards. The EDG building is a seismic Category I structure that meets the HNP design basis requirements for seismic, flood, and wind protection. Supporting systems provide heat, ventilation, and fire protection. These systems protect the equipment from extreme cold and hot temperatures.

   FLEX equipment stored in the EDG building includes: Bay 2A – FLEX ESW Pump (N), associated duplex strainer, discharge manifold, hose, trailer, fittings, and tools; FLEX debris removal equipment, mobile fuel tank and refueling equipment; other FLEX equipment (e.g., lighting, communication equipment, alternate instrument reading equipment), adapters for using NSRC equipment, and EDMP. Bay 2B – FLEX DGs and electrical distribution equipment.

b. RAB and Tank Building – The RAB and the Tank Building provide protected storage for FLEX equipment against all applicable extreme external hazards. The RAB and the Tank Building are seismic Category I structures that meets the HNP design basis requirements for seismic, flood, and wind protection. Supporting systems provide heat, ventilation, and fire protection. These systems protect the equipment from extreme cold and hot temperatures.

   FLEX equipment stored in the RAB and the Tank Building includes: FLEX AFW Pumps, FLEX RCS Pumps, water transfer pump, boron batching, transfer materials; storage boxes for hose, tools, mechanical piping connectors, portable electrical distribution equipment, power cables, portable lighting, and communications equipment.

c. FLEX storage (N+1) Building (FSB) - The FSB is a commercial grade structure located on-site in the owner controlled area west of the protected area. Supporting systems provide heat, ventilation, and fire protection. The FSB is designed to maintain temperature greater than 40°F. FLEX equipment stored in the FSB includes: FLEX ESW pump (N+1) and associated duplex strainer, discharge manifold, hose, trailer, fittings, and tools.

d. Diesel fuel oil storage tank (DFOST) Building. The DFOST building provides protected storage for FLEX equipment against all applicable extreme external hazards. The DFOST building is a seismic Category I structures that meets the HNP design basis requirements for seismic, flood, and wind protection. The design indoor temperature of the DFOST building during normal plant operations range from 60°F to 109°F. FLEX equipment stored in the DFOST building includes: FLEX Fuel Oil Transfer Pump.

During the on-site audit, the NRC staff determined that Duke's strategy, to use permanently pre-staged FLEX DGs, should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. The licensee evaluated the approach for permanently pre-staging two 480V, 830KW FLEX DGs in a single building that is a seismic Category I structure and meets the plant's design-basis with respect to the extreme external hazards. The licensee provided justification for the proposed alternative in the fourth 6-month update [Reference 14]. In its FIP, the licensee stated that FLEX DG connection points are also contained within this structure and the evaluation considered the potential causes of a failure of both FLEX DGs and determined that such a failure is highly unlikely given the design and layout of the FLEX DG installation and the quality of the installed components. The licensee concluded that the evaluation of the FLEX DG installation against the applicable requirements demonstrates that the installation constitutes an acceptable alternative to the NRC endorsed guidance provided in NEI 12-06. The NRC staff found this proposed alternative to be acceptable.

During the audit process, the NRC staff determined that the (N+1) FSB, which houses the (N+1) FLEX ESW pump does not meet the standards of American Society of Civil Engineers (ASCE) 7-10, "Minimum Design Loads for Buildings and Other Structures" and should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. This building was constructed according to the North Carolina building codes. The licensee agreed that this configuration is an alternative to the FLEX storage guidance in NEI 12-06, and implemented the following compensatory measures to limit the potential vulnerability of the pump as an alternative to the NEI 12-06 [Reference 6] guidance. The licensee's alternative approach is discussed in Section 3.14, below.

The NRC staff found the proposed alternative acceptable. Below are additional details on how the FLEX equipment is protected from each of the external hazards.

3.6.1.1 Seismic

As stated in the FIP, and as described in Section 3.6.1, the RAB, Tank, DFOST, and EDG buildings are seismic Category 1 buildings, which are designed to withstand the DBE. As such, these buildings meet the NEI 12-06 [Reference 6] guidance. In its fourth 6-month status report, dated February 23, 2015 [Reference 14], Duke stated that the FSB, which houses the (N+1) FLEX ESW pump, is a commercial metal building constructed to North Carolina building codes.

As discussed in Section 3.6.1, the FSB is not protected against seismic, high winds, and tornados. Duke has implemented compensatory measures, as discussed in Section 3.6.1, in order to address the use of the FSB as an alternative to the guidance described in NEI 12-06 [Reference 6].

3.6.1.2 Flooding

In its FIP, the licensee stated that HNP is a dry site with a nominal plant elevation of 260 feet (ft.) MSL and a maximum water level, due to the probable maximum flood event, of 257.7 ft. MSL. Therefore, the FLEX buildings will not be impacted by external flooding.

As noted in Section 3.5.2 above, the licensee's FHRR showed that some of the flood-causing mechanisms are not bounded by the current plant-specific design-basis hazard. However, the licensee stated that the increased levels did not exceed the flood protection capabilities or did not impact safety-related equipment; therefore no interim actions were planned. The NRC staff finds that the licensee has appropriately provided flood protection for the FLEX equipment and meets the guidance in NEI 12-06.

3.6.1.3 High Winds

As stated in the FIP, and Section 3.6.1 above, the RAB, Tank, DFOST, and EDG buildings are seismic Category 1 buildings and protect against extreme external hazards. Harris UFSAR Section 3.3, "Wind and Tornado Loadings," states that the EDG, RAB and Tank, and DFOST buildings were designed to withstand design wind, tornado wind, and tornado generated missiles.

During the audit process the NRC staff determined that the FSB, which houses the (N+1) FLEX ESW pump does not meet the standards of ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures" and should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. In its fourth 6-month status report, dated February 23, 2015 [Reference 14], the licensee stated that the FSB, which houses the (N+1) FLEX ESW pump, is a commercial metal building constructed to North Carolina building codes. The licensee implemented the following compensatory measures to limit the potential vulnerability of the pump as an acceptable alternative to the NEI 12-06 [Reference 6] guidance. The licensee's alternative approach is discussed in Section 3.14, below.

3.6.1.4 Snow, Ice, and Extreme Cold and Extreme Heat

As described in Section 3.6.1 above, all of the FLEX buildings provide ventilation and heat to control the temperatures inside these buildings. This serves to protect the FLEX equipment from extreme cold and heat. As stated in the FIP, the RAB, tank, DFOST, and EDG buildings are seismic Category 1 buildings and protect against extreme external hazards. Harris UFSAR Section 3.3, "Wind and Tornado Loadings," states that the EDG, RAB and Tank, and DFOST buildings were designed to withstand design wind, tornado wind, and tornado generated missiles. As such, these buildings will function to protect the FLEX equipment from snow and ice. In its fourth 6-month status report, dated February 23, 2015 [Reference 14], the licensee stated that the FSB, which houses the (N+1) FLEX ESW pump, is a commercial metal building constructed to North Carolina building codes.

During the audit process the NRC staff determined that the FSB which houses the (N+1) FLEX ESW pump does not meet the standards of ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures" and should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. The licensee agreed that this configuration is an alternative to the FLEX storage guidance in NEI 12-06 and implemented the following compensatory measures to limit the potential vulnerability of the pump as an acceptable alternative to the NEI 12-06 [Reference 6] guidance. The licensee's alternative approach is discussed in Section 3.14, below.

3.6.2 Reliability of FLEX Equipment

Section 3.2.2 of NEI 12-06 [Reference 6] states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare (i.e., an N+1 capability, where "N" is the number of units on-site.

As previously stated, the NRC staff determined that the FLEX strategy to use permanently pre-staged DGs should be considered an alternative to the guidance in NEI 12-06. The licensee evaluated the alternate strategy regarding the approach for permanently pre-staging the two FLEX DGs and concluded that the installation constitutes an acceptable alternative to the NRC endorsed guidance provided in NEI 12-06. The NRC staff found the proposed alternative to be acceptable.

As previously stated, the NRC staff determined that the FSB, which houses the (N+1) FLEX ESW pump does not meet the standards of ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures" and should be considered an alternative to the guidance in NEI 12-06. The licensee implemented compensatory measures to limit the potential vulnerability of the pump as an acceptable alternative to the NEI 12-06 guidance.

In addition, Duke identified a number of FLEX equipment that would be stored in the EDG, RAB, and Tank Buildings. This equipment includes FLEX ESW Pump (N), associated duplex strainer, discharge manifold, hose, trailer, fittings, and tools; FLEX debris removal equipment, mobile fuel tank and refueling equipment; other FLEX equipment (e.g., lighting, communication equipment, alternate instrument reading equipment), adapters for using NSRC equipment, FLEX AFW Pumps, FLEX RCS Pumps, water transfer pump, boron batching, storage boxes for hose, tools, mechanical piping connectors, portable electrical distribution equipment, power cables, portable lighting, EDMP, and communications equipment.

Based on the number of FLEX equipment identified in the FIP and during the audit review, the NRC staff finds that, if implemented appropriately, the licensee's FLEX strategies include a sufficient number of portable FLEX pumps, FLEX DGs, and equipment for AFW makeup to the SGs, RCS makeup and boration, SFP makeup, and maintaining containment that is consistent with the N+1 recommendation in Section 3.2.2 of NEI 12-06.

3.6.3 Conclusions

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

3.7 Planned Deployment of FLEX Equipment

3.7.1 Means of Deployment

In its FIP, Duke indicated that HNP has a debris removal tractor that is stored in EDG Building Bay 2A, which is robust for the applicable external hazards at HNP. This tractor will ensure transit paths outside of robust structures are clear of snow or other debris for deployment of FLEX equipment. The debris removal tractor will also be used to transport or tow FLEX equipment as necessary. Additional general purpose equipment is stored throughout the site that can also be used for debris removal, including another tractor that is stored in the FSB (N+1). Duke also stated that the FLEX strategies were developed such that most FLEX equipment does not require outdoor transportation. Duke designated outdoor FLEX staging areas and haul paths to support deployment of FLEX strategies. Due to diverse haul path options, no logistical bottlenecks, and the fact that the vast majority of the required response equipment is stored inside the protected area, such pathways do not require specific programmatic monitoring or control. Furthermore, debris removal equipment stored in robust structures is maintained available such that pathways may be cleared in a timely manner following a BDBEE.

Additionally, Duke stated in the third 6-month update [Reference 13] that the EDG building door would be manually operated and a tractor equipped with a front-end loader would be used to relocate FLEX equipment. This would allow for operation of the door during an ELAP event.

During the on-site audit, the NRC reviewed the licensee's evaluation of flooding from sources that are not seismically robust and do not require ac power. Guideline FSG-005, "Initial Assessment and FLEX Equipment Staging" [Reference 42] directs the assessment of specific systems and plant areas necessary to implement FLEX strategies and an assessment of internal flooding. Based upon locations of the postulated internal flooding source pipe breaks, the FLEX guidance and abnormal operating procedures provide instructions for the isolation of any flooding source and recommends measures such as opening doors to allow migration of the flood water to adjacent non-critical plant area.

3.7.2 Deployment Strategies

In its FIP, the licensee stated FLEX equipment and fuel oil transfer travel routes from their storage locations to the point of deployment were evaluated for potential soil liquefaction. During the audit review, the NRC staff reviewed EC 91691, "Analysis of Delivery Path of FLEX Equipment (Item 30)," Revision 1 [Reference 44]. This evaluation concluded that deployment of FLEX equipment and fuel oil delivery following a seismic event would not be hindered.

In its FIP, the licensee stated that appropriate programmatic controls have been established to ensure that FLEX equipment is maintained available for use in site-specific FLEX strategies. Duke's procedures prescribe consideration of deployment paths as part of key safety function impact reviews for tracking unavailability.

Regarding the core cooling strategy, the licensee stated in its FIP, that a FLEX AFW pump is mobilized near the AFW pumps, and each FLEX AFW pump is mounted on a mobile cart to easily transport to its deployed location. Equipment transport paths and hose runs are located inside the Tank Building and RAB, and will be not be compromised during a BDBEE.

The FLEX ESW pump takes suction from an ESW pump bay in the ESW and cooling tower make-up water intake structure and will pressurize an ESW header. FLEX ESW pumps will be deployed to either a location near the selected ESW train (primary strategy) or near the auxiliary reservoir (alternate strategy). Hose runs for connecting the ESW pump are outdoors or inside the ESW and cooling tower make-up water intake structure, which is robust to applicable external hazards. For outdoor staging locations and hose runs, the debris removal vehicle will clear debris to enable deployment of a FLEX ESW pump and hoses.

Regarding the RCS boration strategy, in its FIP, Duke stated that both of the FLEX RCS pumps are stored in locations of the Tank Building that are protected from all applicable extreme external hazards. A FLEX RCS pump is mobilized at the charging pump rooms and is aligned to FLEX connection points on the charging/safety injection pumps piping. Each FLEX RCS pump is mounted on a mobile cart to enable movement to its deployed location. Equipment transport paths and hose runs are located inside the Tank Building and RAB and will not be compromised during a BDBEE.

Regarding the electrical strategy, in its FIP, the licensee stated that power is provided by one of two pre-staged FLEX DGs located in the EDG building bay 2B, which is robust to the applicable extreme external hazards at HNP. The ac power from either FLEX DG can be routed through an attached output breaker to one of two patch panels via portable cables to a 480V to 6.9 kV transformer. Power from the 6.9 kV transformer is routed through the FLEX cabling located within safety-related conduit into the RAB. The cables are routed to medium voltage termination boxes near 6.9 kV emergency busses 1A-SA and 1B-SB. Either termination box can be connected via portable cables to one of two FLEX temporary power units to power one of two 480V safety-related buses and FLEX MCC-RAB. In this manner, the FLEX DGs, associated output devices, and cabling are protected from all applicable extreme external hazards while remaining physically disconnected from the plant's installed electrical distribution system during normal plant operation. Support equipment for each FLEX DG includes a booster fan, a manual balance damper, flexible duct, and connectors, which are stored in the EDG building bay 2B. Equipment transport paths and cable runs are internal to the building and protected from all applicable extreme external hazards.

Regarding the SFP cooling strategy, in its FIP, the licensee stated that SFP makeup water consists of using a FLEX ESW pump to pressurize an ESW header, which can supply make-up water through the SSE Fire Protection (FP) header. Connection points on SSE FP hose stations will be utilized by the strategy.

For the Phase 3 strategies, in its FIP, Duke stated that if possible, SAFER equipment will be delivered to staging area B, which is on the north side of the HNP site. A helicopter landing area is near staging area B to enable helicopter transport of equipment, if necessary. Staging area C (Siler City Airport; 41 miles away by driving) or staging area D (Raleigh Executive Jetport; 17 miles away by driving) may also be used. If road transportation is not available from the airports, helicopter transportation options are delineated in the response plan.

3.7.3 FLEX Connection Points

3.7.3.1 Mechanical Connection Points

The licensee stated in its FIP that FLEX connections are installed on various plant systems. Primary and alternate connections are available for each system used for FLEX. These connection points are protected from the applicable extreme external hazards as discussed in each area.

Primary and Alternate Connection Points for Core Cooling

As described in the HNP FIP, Duke will supply AFW to the SGs using one of two portable FLEX AFW pumps. Primary and alternate suction connections for these pumps are located on the A and B Motor-Driven AFW Pump suction strainers. Primary and alternate discharge connection points for the FLEX AFW Pumps are downstream of the installed motor-driven AFW pumps. All connection points are inside the RAB, which is a Seismic Category I structure that is protected from all applicable extreme external hazards. The AFW A and B train do share common piping depending on the chosen flow paths. However, operators can configure the system such that the primary and alternate connection points would provide independent flow paths to individual SGs. Based on the design and location of the primary and alternate AFW connection points, as described in the FIP and Harris UFSAR, at least one of the connection points should be available to support core cooling via a portable pump during an ELAP caused by a BDBEE, consistent with NEI 12-06 [Reference 6], Section 3.2.2 and Table D-1.

In its FIP, the licensee stated that if the CST is no longer available as a source for AFW, Duke may use water from the auxiliary reservoir via an ESW Header. For this approach, Duke would use a FLEX ESW pump to pressurize the header and enable flow. The primary suction path is established by routing hoses into the an ESW pump bay, which is inside the ESW and cooling tower make-up water intake structure which is a seismic Category I structure that is protected from all applicable extreme external hazards. The primary and alternate discharge connections for the FLEX ESW pump are on the normal discharge piping for the installed A and B ESW pumps. These connections are also in the ESW and Cooling Tower Make-up Water Intake Structure. Once the ESW header is pressurized, it may then be aligned to AFW piping to supply water to the TDAFW pump or the FLEX AFW pump using installed system valves.

As described in the Harris UFSAR Table 3.2.1-1, both the AFW and the credited portions of the ESW systems are both safety-related, seismic Category I systems. The AFW system is located in the RAB (a seismic Category I building that is tornado wind and missile protected). The ESW piping is located in the RAB, and the ESW and Cooling Tower Make-up Water Intake Structure, described in Harris UFSAR Table 3.2.1.-1, is a seismic Category I structure that is tornado wind and missile protected. Portions of the ESW piping are routed underground from the ESW and Cooling Tower Make-up Water Intake Structure to the RAB underground. As detailed in Harris UFSAR Section 9.2.1, the underground sections of piping are buried below the maximum missile penetrating depth. Based on the location and protection, the AFW and ESW systems should be available following an ELAP event.

RCS Inventory Control - MODES 1-4

The licensee stated in its FIP, that the FLEX strategy for RCS makeup relies on connecting one of two FLEX RCS pumps to deliver borated water. The FLEX RCS pump primary and alternate connections are located in the suction and discharge lines of the CSIPs in the charging pump rooms. Operators can configure the system such that the primary and alternate connection points would provide independent flow paths to the reactor vessel. As described in the Harris UFSAR Table 3.2.1-1, both the SI and the credited portions of the CVCS systems are both safety-related, seismic Category I systems. These connection points are located inside the RAB, which is a seismic Category I structure that is protected from all applicable extreme external hazards.

SFP Cooling

The SFP hose connections are discussed Section 3.3.4.1.1. Also, the SFP cooling strategy relies on the ESW connection described above.

3.7.3.2 Electrical Connection Points

According to its FIP, the licensee has made modifications to facilitate the electrical connections required to repower the vital battery chargers from the FLEX DGs. For Phase 2, this will be accomplished by routing power from one of the two pre-staged FLEX DGs located in the EDG Building Bay 2B, which is a Seismic Category I structure that is protected from all applicable extreme external hazards. Power from the FLEX DGs can be routed through an attached output breaker to one of two patch panels also located in EDG Building Bay 2B, via portable cables to a 480V-to-6.9kV transformer. The two patch panels provide primary and alternate connection points. Power from the 6.9 kV transformer is routed through the FLEX cabling located within safety-related conduit into the RAB and routed to medium voltage termination boxes near 6.9 kV Emergency busses 1A-SA and 1B-SB. The termination boxes can be connected via portable cables to one of two FLEX temporary power units to power the one of two 480V safety-related buses and FLEX MCC-RAB. Connecting the FLEX DGs in this manner ensures that they are physically disconnected from the plant's installed electrical distribution system during normal plant operation. For phase 3, a NSRC FLEX DG can be connected via the 480V patch panels located in the EDG Building Bay 2B. The connection points on the patch panels are compatible with the NSRC FLEX DG cables.

3.7.4 Accessibility and Lighting

In regards to accessibility, FSG-005, Initial Assessment and FLEX Equipment Staging, Revision 0, Attachment 4 [Reference 42] provides the initial evaluation/establishment of plant site and building access control and determination of accessibility for FLEX implementation. Included is the possibility that some security doors may be required to be blocked open. FSG-005, Initial Assessment and FLEX Equipment Staging, Revision 0 [Reference 42] also directs the staging of portable lighting.

As previously stated, HNP has a debris removal tractor that is stored in EDG Building Bay 2A. This tractor will ensure transit paths outside of robust structures are clear of snow or other debris for deployment of FLEX equipment.

In regards to lighting, in its FIP the licensee stated that upon a loss of all ac, the MCR lighting will be powered by the 125-volt non-nuclear safety battery, if it is available. Light-equipped safe-shutdown hard hats and hard-hat-mounted flashlights are staged in the MCR. Portable battery-powered lighting units are stored in the RAB near the MCR. Flashlights, batteries, and additional hard hats with headlamps are located in the auxiliary control panel tool locker in the RAB one elevation below the MCR. Additionally, HNP has additional portable battery-powered lighting units stored in the EDG building bay 2A.

Installed self-contained dc emergency lighting units automatically illuminate access and egress routes to and from all fire areas and in areas that must be manned for safe-shutdown. This emergency lighting has a coping time of 8 hours. Supplemental battery powered lighting is available in multiple FLEX storage areas to support operator mobility and equipment deployment. When deploying equipment outdoors, the FLEX debris removal equipment, as well as the trailer mounted FLEX ESW pumps, are equipped with high-powered integrated lighting.

Upon energizing the FLEX electrical distribution system, MCC 1 provide power to safety-related lighting in the MCR and throughout the power block. The FLEX electrical distribution network can then be used to power auxiliary equipment and recharge portable lighting units.

3.7.5 Access to Protected and Vital Areas

During the audit process the NRC staff requested information on how access to other areas of the plant would be accomplished. The licensee stated on site personnel have access to locked FLEX buildings. In addition, as stated previously, FSG-005, Attachment 4 [Reference 42] directs the initial evaluation and establishment of plant site and building access control.

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee stated that diesel fuel oil (DFO) can be transferred from one of two Unit 1

DFOSTs using an air operated FLEX (DFO) transfer pump located in the 2A transfer pump room. Suction can be taken from either of the Unit 1 DFOSTs using a FLEX hose. Fuel can be transferred at a rate of 30 gpm to a trailer mounted fuel oil tank for distribution to diesel-driven FLEX equipment.

Another strategy to move DFO to the trailer-mounted tank utilizes the FLEX electrical distribution system, so that one of the Unit 1 DFO transfer pumps can be powered from MCC 1A35-SA or 1B35-SB. Repowering one of the Unit 1 DFO transfer pumps will facilitate the transfer of fuel from one of the DFOSTs to the trailer-mounted tank via a hose installed at the pump vent line. The two DFOSTs can be tied together using FLEX hose such that only one fuel oil transfer pump needs to be energized. The DFO can also be transferred directly from the DFOST(s) to the EDG building using the Unit 1 fuel oil transfer pumps energized by the FLEX electrical distribution system. This method will facilitate transfer of DFO from one of the DFOSTs to its associated day tank in the Unit 1 side of the EDG Building. An existing flanged pipe connection on the 3-inch drain line of each Unit 1 EDG Day Tank can be disassembled and a hose connected to the flange. The hose can be routed over to the Unit 2 area of the EDG building or outside to the trailer mounted tank and fuel can be gravity drained from the day tank.

During the onsite audit, the licensee provided EC 91683, "Diesel Fuel Oil Transfer Evaluation for NTT 4.2 (FLEX)," Revision 0 [Reference 49]. This evaluation showed the total fuel consumption rates for all of the diesel-powered FLEX equipment anticipated to be operating during a BDBEE is estimated to be 190 gallons/hr. The minimum amount of DFO available on site in the two DFOSTs and EDG Day Tanks at any given time is calculated to be approximately 202,000 gallons. Therefore, the duration before any offsite phase 3 fuel delivery is required will be at least 44 days. Guideline FSG-005, Initial Assessment and FLEX Equipment Staging, Revision 0, Attachment 3, 'Establishing Diesel Fuel Oil Sources and Refueling Capability,' [Reference 42] provides the instructions for fueling the FLEX equipment.

The NRC staff's review of the licensee's maintenance and testing of FLEX equipment is documented in SE Section 3.13.

3.7.7 Conclusions

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, 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 Strategic Alliance for FLEX Emergency Response Plan (SAFER)

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

There are two NSRC's strategically located across the country in Memphis, Tennessee and Phoenix, Arizona. 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 from the NSRC. The primary location for HNP is Memphis.

The SAFER Response Plan for HNP, contains (1) SAFER control center procedures, (2) NSRC procedures, (3) logistics and transportation procedures, (4) staging area procedures, which includes 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.

By letter dated September 26, 2014 [Reference 26], the NRC staff issued its staff assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the NRC 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 [Reference 6] guidance; therefore, the NRC 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.

3.8.2 Staging Areas

The licensee stated in its FIP that If possible, SAFER equipment will be delivered to staging area B, which is on the north side of the HNP site. A helicopter landing area is near staging area B to enable helicopter transport of equipment, if necessary. Staging area C (Siler City Airport; 41 miles away by driving) or staging area D (Raleigh Executive Jetport; 17 miles away by driving) may also be used. If road transportation is not available from the airports, helicopter transportation options are delineated in the response plan.

Guideline FSG-005, Initial Assessment and Flex Equipment Staging, Revision 0, [Reference 42] provides direction for interface with the NSRC. The first arriving equipment will be delivered to the site within 24 hours from initial contact and remaining equipment will be delivered within 72 hours from initial contact.

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, 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 licensee performed an analysis of the effects that an ELAP would have on various room and area temperatures. During the onsite audit, NRC staff reviewed Calculation HMP-M/FLEX-0012, "Harris Reactor Auxiliary Building Extended Loss of ac Power FLEX Response", Revision 0 [Reference 72]. The calculation uses the GOTHIC version 8 computer program (Generation of Thermal-Hydraulic Information for Containments). The calculation developed a single model for the Harris RAB elevations 190', 216', 236', 286', and 305'. The model includes the MCR. The model also included stairways along with doors. The model was run to determine the temperature response up to seven days.

Three cases were run to determine area temperatures following an ELAP. The first case was based on extreme summer input. The summer solar loading was based on an extreme site outdoor temperature in addition to accounting for energized equipment within the RAB that would add heat to the rooms during the event. This case was run without any operator actions taken, e.g. propping doors open and/or installing portable fans. The second case considered extreme winter conditions. As with case 1, this scenario was run without any operator actions. The third case is based on case 1 with the addition of operator actions to prop doors open and use of portable fans in strategic locations to create ventilation paths through the RAB. The timing for when doors would be propped open was factored in the calculation.

A fourth case was run to model battery room hydrogen generation. This model included opening doors to establish ventilation paths.

The licensee states that the GOTHIC model showed the maximum temperature in the MCR at the end of seven days to be 104.4°F. Procedure FSG-005, 'Initial Assessment and FLEX Equipment Staging,' Revision 0 [Reference 42] directs opening doors to MCR and the installation of temporary fans in response to a loss of control room ventilation and provides a sketch of the fan locations. Based on temperatures remaining below 120°F (the temperature limit, as identified in NUMARC-87-00, for electronic equipment to be able to survive indefinitely), the NRC staff finds that the equipment in the MCR will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

The licensee's Phase 1 core cooling FLEX strategy relies on the TDAFW pump as the motive force for providing water to the SGs. The Calculation HMP-M/FLEX-0012 [Reference 72] determined the temperature in the TDAFW pump area would reach a maximum temperature of 117.5°F after a period of 7 days. During the audit process, the licensee provided Engineering Change (EC) 91690, "Loss of HVAC [Heating, Ventilation, and Air Conditioning] Analysis," Revision 0, Attachment T01 [Reference 77]. This evaluation stated the TDAFW pump would be able to function for at least 24 hours in the BDBEE environment of 117.5°F. The NRC staff also noted the discharge from TDAFW pump provides cooling water to the turbine driver's governor oil and lube oil coolers. While onsite, the NRC staff walked down the TDAFW pump area noting the TDAFW pump is located in a relatively open space. Additionally, FSG-005 [Reference 42] has steps to add supplemental ventilation as necessary to areas where FLEX operations are performed.

The licensee's Phase 2 core cooling and RCS makeup strategy rely on portable FLEX pumps. Both pumps will be staged in the RAB in the vicinity of the TDAFW pump so they will be subject similar maximum temps during a BDBEE. During the audit, the licensee provided pump data sheets, EC 91681, Attachment Z01, "FLEX AFW Pump Data Sheet," Revision 2 and EC91680, Attachment Z08, "FLEX RCS Pump Data Sheet," Revision 3, that listed a maximum ambient operating temperature of 120°F for both pumps.

The licensee states the battery room temperature will reach 92°F in 24 hours and 115°F in 7 days during an extreme heat event. The NRC staff walked down the vital battery rooms and reviewed Calculation HNP-M/FLEX-0012 [Reference 72] and EC 91690 [Reference 77] to confirm the adequacy of the battery room ventilation. The licensee plans to restore normal battery room ventilation when the batteries are charging during Phase 2 and 3. The fans would draw air as designed through the battery rooms and the room temperatures would trend toward the ambient air temperature of the auxiliary building interior. As a result of its review, the NRC staff finds the equipment in the MCR, RAB, and battery room, will not be adversely impacted and should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP.

3.9.1.2 Loss of Heating

In response to audit questions from the NRC regarding the effects on FLEX due to a loss of heat tracing and severe cold weather conditions, the licensee stated that an evaluation was performed in EC 91690, Attachment TO1, "Loss of HVAC Analysis," Revision 0 [Reference 77] to determine the expected heating and cooling of select areas of the plant during a BDBEE resulting in loss of heating, ventilation, and air conditioning (HVAC). The analysis provided the maximum and minimum expected temperatures, equipment limitations, expected personnel locations, and recommended strategies to mitigate overheating or overcooling of equipment and personnel. This calculation analyzed the loss of heat trace for various plant areas and found the Main Steam Tunnel, RWST and RAB 261' level were the areas that needed mitigating action for extreme cold weather. Guideline FSG-005, "Initial Assessment and FLEX Equipment Staging," Revision 0 [Reference 42] contains direction to initiate heat tracing and freeze protection. The FSG contains guidance to block ventilation louvers and other Main Steam Tunnel openings, to monitor temperatures and to set up portable FLEX heaters, if necessary, to maintain temperature. The RWST instrumentation and small bore piping can be protected by supplying FLEX power to the one of two redundant freeze panels and portable FLEX heat cable and insulating blankets can be used to the RWST piping. Furthermore, the FSG contains guidance to monitor the BAT temperature (located on the 261' level with portable FLEX equipment and recirculate and/or flush the BAT, and/or place temporary heaters near boric acid piping to prevent boron precipitation.

Additionally, FSG-005 [Reference 42] contains precautions to drain and disconnect hoses and or provide supplemental heating to prevent freezing during extreme cold conditions. The FSG also directs operators to add cold weather fuel additive to FLEX equipment. Guideline FSG-005 [Reference 42] also contains direction to establish a recirculation path for the FLEX ESW Pump, the only pump operated outside. A recirculation path with continuous flow will help to prevent freezing of FLEX components. Furthermore, the licensee provided EC 91682, Attachment Z31, "Engineering Data Sheet for FLEX ESW PMP A," Revision 0 and EC 91682, Attachment Z32, "Engineering Data Sheet for FLEX ESW PMP B," Revision 0 that showed the A and B FLEX ESW Pumps operating temperature range down to -9°F. Guideline FSG-011, "Alternate SFP Makeup and Cooling," Revision 0, [Reference 41] provides guidance to use the hose stations on the 261' level of the RAB when extreme cold temperatures exist. Hoses from these hose station not routed outside RAB and Fuel Building which will aid in preventing the hoses from freezing.

The licensee states the lowest battery room temperature reached in 24 hours is about 65°F and at 7 days the temperature is expected to lower to about 60°F. Based on the rate of temperature decrease, the NRC staff finds it reasonable that the battery rooms will not be impacted by the loss of heating.

3.9.1.3 Hydrogen Gas Accumulation in Vital Battery Rooms

The NRC staff reviewed Calculation, HNP-M/FLEX-0012 [Reference 72] and EC 91690 [Reference 77] to verify that hydrogen gas accumulation in the 125 Vdc Vital Battery rooms will not reach combustible levels while HVAC is lost during an ELAP as a result of a BDBEE. Duke's analysis considered hydrogen gas generation rates provided by the battery manufacturer (C&D Technologies) during a float charge and worst-case maximum temperatures (120°F). Duke calculated that the maximum hydrogen concentration of 0.27 percent by volume occurs at 168 hours into the event.

Based on its review of the analysis, Duke concluded that hydrogen accumulation in the 125 Vdc Vital Battery rooms should not reach the combustibility limit for hydrogen (4 percent) or the plant limit (1 percent) during an ELAP as a result of a BDBEE since it is assumed that battery room doors will be open and power will be restored to the vital battery rooms HVAC systems within the calculated times before hydrogen gas accumulation reaches 1 percent in the 125 Vdc Vital Battery rooms. The NRC staff did not identify any discrepancies with the analysis.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

Personnel activity will occur in the MCR to support the FLEX strategies and monitoring plant conditions. Case 1 indicates the MCR temperature would be 104.4°F at the end of 7 days with doors closed. As temperatures rise in this and adjacent areas, cooling fans may be employed and doors may be propped open to prevent heat stress as warranted. As personnel will continuously inhabit the MCR, any fans used for personnel cooling will also maintain the room temperatures to below that of the instrument design temperatures.

3.9.2.2 Spent Fuel Pool Area

Duke performed Calculations HPN/FLEX-0013, "Harris Fuel Handling Building Extended Loss of AC Power FLEX Response," Revision 0 [Reference 78] and HPN/FLEX-0016, "Harris Fuel Handling Building Alternative Cooling," Revision 0 [Reference 95] to address the environment in the FHB. Duke analyzed the FHB with doors closed and opening some doors to provide ventilation paths. Plant staff will deploy hoses and stage spray nozzles as part of the FLEX strategies for providing makeup and cooling to the spent fuel pool. FSG-011, “Alternate SFP makeup and cooling”, Revision 0, Step 11, [Reference 41] provides guidance that ventilation paths need to be established within 2 to 3 hours following ELAP. Guideline FSG-005, “Initial Assessment and FLEX Equipment Staging”, [Reference 42] provides guidance for deployment of hoses and spray nozzles.

3.9.2.3 Other Plant Areas

Duke plans to make a portable command trailer available which is equipped with HVAC. The command trailer may also be used to support personnel comfort during events resulting in extreme temperatures. In addition, HNP has portable HVAC equipment which can be utilized to establish a personnel comfort station in any enclosed structure. This equipment provides defense in depth for extreme temperature events.

Portable fans will be available for local forced ventilation as needed. Fans will be powered by power buggies deployed per the FLEX electrical distribution strategy.

Duke’s calculation indicates the TDAFW pump area temperatures will be less than 120°F. Duke also performed an operability analysis of the TDAFW pump and turbine operating in this ambient air temperature. Given this information, the TDAFW pump should be able to function during an ELAP and that operators will be able to manually operate it.

Duke’s calculations were evaluated to determine if any plant area temperatures are expected to exceed fire suppression actuation setpoints. Areas where the potential exists for area temperatures near or exceeding the fire suppression actuation setpoints, area temperatures will be monitored, guidance to mitigate the area temperature using supplemental ventilation will be provided. If necessary, isolation of fire suppression to the affected area to prevent negative impact on equipment required to support FLEX strategies.

Guideline FSG-005, Step 11 initiates providing alternate HVAC cooling and ventilation. Attachment 7 to FSG-005, “FLEX HVAC Cooling and Ventilation”, Revision 0, [Reference 42] provides guidance for establishing ventilation pathways as required. In addition existing plant procedures addressing heat stress mitigation actions provide reasonable personnel will be able to implement FLEX mitigation activities.

The following plant procedures provide heat stress mitigation actions: AD-HS-ALL-0106, “Heat and Cold Stress, Revision 2, AP-300, “Severe Weather Response”, AP-301, “Seasonal Weather Preparations, and SAF-SUPS-00032, “Heat Stress.”

3.9.3 <u>Conclusions</u>

Based on the evaluation above, 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, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.10 Water Sources

3.10.1 Steam Generator Make-Up

Phase 1

The licensee stated in its FIP that the CST is the water source to the TDAFW pump. The CST is located in the Tank Building and is the initial credited source of AFW for the FLEX core cooling strategy. As described in the Harris UFSAR Table 3.2.1-1, the Tank Building is a seismic Category I structure and the CST is a seismic Category I Tank. However, there is a non-seismic, condensate return nozzle that taps in to the tank approximately two-thirds up the tank. Failure of the nozzle during a seismic event limits the available volume to 62 percent capacity (238,000 gallons), which would be sufficient for at least 21 hours. Normal operating level for the tank is 80 percent to 91 percent so during a non-seismic event, the available CST capacity is 80 percent (324,800 gallons) and would be a sufficient supply of water for at least 36 hours.

In EC 91684, "Calculation of Available Volume from Condensate Storage Tank (Seismic and Normal), Revision 0, [Reference 50], Duke determined the available CST water capacity available to support the FLEX core cooing strategies during the applicable ELAP event.

If implemented appropriately and consistent with the FIP, the licensee should have water sources available during the Phase 1 core cooling strategies for SG inventory makeup. The licensee's sequence of events timeline, as documented in its FIP, notes that the RWST or RMWST can be aligned for service within 18 hours. In addition, the licensee's strategy should provide sufficient time for operators to begin deploying and staging Phase 2 FLEX equipment.

Phase 2

In its FIP, the licensee stated that the CST water supply is credited to be available for at least 36 hours for non-seismic events. For a seismic event, the CST is credited to be available for at least 21 hours. As CST depletion approaches, an ESW header may be pressurized by one of two portable FLEX ESW pumps taking suction from an ESW pump bay in the ESW and cooling tower make-up water intake structure.

Guideline FSG-003, "FLEX Low Pressure Feedwater," Revision 0, [Reference 52] outlines instructions for feeding the SGs using the FLEX AFW Pump. This procedure contains guidance to use the RMWST or RWST if ESW is not pressurized. As described in Harris UFSAR Table 3.2.1-1, both the RWST and RMWST are seismic Category I tanks and are located in the Tank Building. The RMWST has an available credited volume of approximately 28,500 gallons. As described in the licensee's FIP, the RWST is robust to all applicable extreme external hazards at HNP, except for wind-generated missiles. For all other events, the available credited volume is 407,000 gallons. For a tornado missile event, the available credited volume is approximately 187,000 gallons. This volume accounts for the presence of a concrete wall that protects the lower portion of the RWST from wind-generated missiles. If CST depletion is anticipated to occur prior to ESW header pressurization, Duke will use inventory directly from the RMWST and the RWST for AFW, until ESW is pressurized and aligned to AFW.

As described in the licensee's sequence of events, the FLEX ESW pump will not be staged and available until 35 hours into the event which supports the non-seismic CST usage timeline. As a measure of defense in depth for a seismic event, the licensee created FSG-006, "Alternate CST Makeup," Revision 0 [Reference 51] which contains procedures for making up to the CST using any available (credited or non-credited water source). Furthermore, the FLEX AFW Pump can be connected and available 12 hours into the event.

Regarding access to the auxiliary UHS during extreme cold, Duke's analysis concluded that extreme icing is not a concern for the auxiliary reservoir based on the design of the ESW system drawing water from more than 10 feet below the low water level, barriers that prevent from ice entering the ESW intake bays, and historical meteorological data indicating that extreme icing sufficient to overcome these design features is unlikely.

During the onsite audit, NRC staff walked down the FLEX ESW pump placement and deployment paths and determined a substantial amount of surface icing would be required to affect the suction of the FLEX ESW pump. Additionally, FSG-005, "Initial Assessment and FLEX Equipment Staging", Revision 0 [Reference 42] establishes a recirculation path for the FLEX ESW pump to ensure constant flow and prevent freezing. Alternatively, the FLEX ESW pump suction hose can be placed directly in the auxiliary reservoir.

If the traveling screens are blocked, the FLEX ESW pump suction may be placed directly into the auxiliary reservoir. Duke will have an ESW header pressurized and aligned to the AFW system at Harris within 36 hours of the start of the event. The auxiliary reservoir, the ESW header, and permanently installed piping relied upon for the FLEX strategy are robust to all applicable extreme external hazards at HNP. The auxiliary reservoir will provide a sustained water supply with essentially indefinite capacity.

If implemented appropriately and consistent with the FIP, the licensee should have a reliable water source available during the Phase 2 reactor core cooling strategies for SG inventory makeup. In addition, the licensee's strategy should provide sufficient time for operators to begin deploying and staging Phase 3 FLEX equipment.

Phase 3

The Phase 3 core cooling strategy continues to use the SGs as the heat sink using water supplies discussed in Phase 2 above. If necessary, the FLEX AFW, and ESW pump can be replaced by pumps from the NSRC. Required adapters to support connection of the NSRC pumps are stored in the RAB and EDG Building Bay 2A. Duke will receive water treatment equipment from the NSRC that can be used to provide clean water for long-term core cooling. If implemented appropriately and consistent with the FIP, the licensee should have an adequate source of water available during the Phase 3 core cooling.

3.10.2 Reactor Coolant System Make-Up

Phase 1

The licensee stated in its FIP that with no intervention for RCS inventory, reflux cooling could occur at 14.8 hours and core uncovery at 44.9 hours. The greatest boration requirements occur at the end of the core cycle, and Duke calculated that k_{eff} remains below 0.99 for 20 hours at 350°F. Based on these evaluations, Phase 1 actions are not required for RCS inventory and reactivity control.

If implemented appropriately and consistent with the FIP, the licensee's approach should conserve RCS inventory to preclude the necessity for RCS system makeup during Phase 1.

Phase 2

The licensee stated in its FIP that to provide a means of RCS makeup, as well as maintaining subcriticality via RCS boration, an electric-powered FLEX RCS pump will be mobilized at the CSIP rooms. The sources of borated water will be from the BAT, ASI Tank, or RWST.

As described in the Harris UFSAR Table 3.2.1-1, the BAT is a seismic Category I tank. The BAT is located in the RAB and is the credited source of water for RCS boration. The BAT is robust to all applicable extreme external hazards at HNP. The available credited volume is 24,150 gallons and the normal boron concentration is 7,000 ppm to 7,750 ppm. This volume and boron concentration is sufficient for the initial boration of 4,800 gallons. Additional BAT inventory may be diluted and used for long-term RCS inventory management, as necessary. A description of the RWST is provided in Section 3.10.1 in the Phase 2 Section. As another backup source, the ASI Tank is located in the RAB and contains 33,600 gallons of at least 3,800 ppm borated water can be used for RCS makeup. The ASI tank is not considered seismically robust, but can be used if available.

If implemented appropriately and consistent with the FIP, the licensee should have a sufficient source of water available during the Phase 2 to maintain RCS inventory to maintain natural convection cooling and control reactivity in the core.

Phase 3

The licensee stated in its FIP that HNP can use the NSRC water treatment equipment effluent to dilute the following tank inventories of a high boron concentration for makeup to the RWST: BAT, ASI tank, and recycle monitor tanks 1 and 2B. Borated makeup water can be mixed in the in the recycle monitor batch tank for transfer to the RWST. Inventory from the SFP can be transferred to the RWST as well.

If implemented appropriately and consistent with the FIP, the licensee should have a sufficient source of water available during the Phase 3 to maintain RCS inventory to maintain natural convection cooling and control reactivity in the core.

3.10.3 Spent Fuel Pool Make-Up

Phase 1

The licensee stated in its FIP that no actions are anticipated during Phase 1 for SFP makeup because the time to fuel uncovery (greater than 72 hours) is sufficient to enable deployment of Phase 2 equipment. Duke will monitor SFP water level using SFP level instrumentation. If necessary, SFP makeup can be provided by gravity drain from the RWST. A description of the RWST is provided in Section 3.10.1.

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phases 1 to maintain SFP cooling.

Phase 2

The licensee stated in its FIP that the strategy to provide SFP makeup water consists of using a FLEX ESW Pump to pressurize an ESW header, which can supply make-up water through the SSE FP header. Connection points on SSE FP hose stations will be utilized by the strategy. This will provide a makeup rate in excess of 250 gpm. Based on design-basis heat loading, 105-gpm total flow is required to maintain SFP levels (90 gpm for the combination of pools A and B; 15 gpm for pool C).

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phase 2 to maintain SFP cooling.

Phase 3

In its FIP, the licensee stated that HNP will receive water treatment equipment from the NSRC to ensure a long-term source of clean water for SFP cooling. If necessary, the FLEX ESW pump can be replaced by a pump from the NSRC. Required adapters to support connection of the NSRC pump are stored in the RAB and EDG Building Bay 2A.

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phase 3 to maintain SFP cooling.

3.10.4 Containment Cooling

Phase 1

The licensee stated in its FIP that Duke performed GOTHIC analysis to determine the temperature and pressure increase resulting from an ELAP during a BDBEE. Duke concluded that no actions were required as part of Phase 1, aside from monitoring containment pressure.

Phase 2

The licensee stated in its FIP that Duke performed GOTHIC analysis to determine the temperature and pressure increase resulting from an ELAP during a BDBEE. Duke concluded that no actions were required as part of Phase 2, aside from monitoring containment pressure.

Conservatively, Duke's FLEX strategy for containment integrity will start containment cooling within 48 hours of the ELAP. When the ESW header is pressurized using a FLEX ESW pump, a containment air cooler can be placed into service. From the time the fan cooler is started, 48 hours into the event, the fan cooler immediately begins removing more heat than is being added.

Heat removal for containment cooling can be provided by using the auxiliary reservoir and a FLEX ESW pump to pressurize an ESW header. The auxiliary reservoir and the ESW piping are either located in seismic Category I structures or are robust to all applicable extreme external hazards. When an ESW header is pressurized a containment air cooler can be placed into service. Four containment air coolers, two per train, are located inside containment, which is a seismic Category I structure that is protected from all hazards. A containment air cooler is qualified to perform its design function at the limiting environmental conditions of containment, which bounds the calculated pressure and temperature resulting from the postulated event.

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phase 2 to maintain containment cooling.

Phase 3

The licensee stated in its FIP that the Phase 3 strategy for containment integrity is to maintain the Phase 2 approach. If necessary, the FLEX ESW pump can be replaced by a pump from the NSRC. Required adapters to support connection of the NSRC pump are stored in the RAB and EDG building bay 2A. The licensee also stated in its FIP, that containment spray functionality is not required to support FLEX strategies.

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phase 3 to maintain containment cooling.

3.10.5 Conclusion

Based on the above, if implemented appropriately and consistent with the FIP, the licensee should have water sources available for SG makeup, RCS makeup, SFP makeup, and containment cooling following a BDBEE consistent with NEI 12-06, 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 a BDBEE occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. When the BDBEE occurs with the plant at power, the mitigation strategy initially focuses on the use of a steam-driven TDAFW 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. By letter dated June 29, 2016 [Reference 94], the licensee stated that following a full core offload to the SFP, about 61.9 hours would be required to implement makeup before boil-off results in the water level in the SFP dropping far enough to uncover the fuel assemblies. Duke 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 the TDAFW pump (which typically occurs when the RCS has been cooled below about 300 °F), another strategy must be used for decay heat removal. On September 18, 2013, NEI submitted to the NRC a position paper entitled "Shutdown/Refueling Modes" [Reference 27], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 28], the NRC staff endorsed this position paper as a means of meeting the requirements of the order.

The position paper provides guidance to licensees for reducing shutdown risk by incorporating FLEX equipment in the shutdown risk process and procedures. Considerations in the shutdown risk assessment process include maintaining necessary FLEX equipment readily available and potentially pre-deploying/pre-staging equipment to support maintaining or restoring key safety functions in the event of a loss of shutdown cooling. The NRC staff concludes that the position paper provides an acceptable approach for demonstrating that the licensees are capable of implementing mitigating strategies in shutdown and refueling modes of operation. In the second 6 month update [Reference 12] and again in its FIP, Duke committed to following the guidance in the NEI position paper entitled "Shutdown / Refueling Modes".

In its FIP, Duke described the FLEX strategy for core cooling during Modes 5 and 6 as follows:

- When the RCS is depressurized (i.e., SGs not available as heat sink), the RWST will be used to gravity feed the RCS as the Phase 1 core cooling strategy. For Phase 2, a FLEX AFW Pump can be used to inject water to the RCS from the RWST. The pump suction can be aligned to a connection point on either A or C Charging/Safety Injection Pumps (CSIPs) suction header. The pump discharge can be aligned to a connection point on either A or B CSIPs discharge header. Injection flow will be manually controlled to maintain core exit thermocouple temperature indications stable and ensure effective use of available RWST inventory. If necessary, the FLEX AFW pump can be replaced by a pump from the NSRC for Phase 3. Required equipment to support connection of the NSRC pump is stored in the RAB and EDG Building Bay 2A.

- When the RCS is pressurized, or capable of being pressurized, SG inventory will be used to remove heat by steaming through SG PORVs for Phase 1. The Phase 2 and 3 strategy is the same as utilized for an ELAP occurring with the plant in Modes 1 through 4.

Based on the above, 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, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.12 Procedures and Training

The licensee stated in its FIP that the inability to predict actual plant conditions that require the use of BDBEE equipment makes it impossible to provide specific procedural guidance. As such, the FSGs provide guidance that can be employed for a variety of conditions. The FSGs, to the extent possible, provide pre-planned FLEX strategies for accomplishing specific tasks in support of EOPs and Abnormal Operating Procedures (AOPs). The FSGs are used to supplement the existing procedure structure that establishes command and control for the event. Procedural interfaces were incorporated into ECA-0.0, "Loss of All AC Power," [Reference 40] to the extent necessary to include appropriate reference to FSGs and provide command and control for the ELAP.

Training programs and controls have been established to assure personnel proficiency in the mitigation of BDBEE is developed and maintained. The Systematic Approach to Training (SAT) process was utilized to evaluate, develop and implement training for applicable personnel.

Initial training has been provided and continuing periodic training will be provided to site emergency response leaders on BDBEEs emergency response strategies and implementing guidelines. Personnel assigned to direct the execution of mitigation strategies for BDBEEs have received the necessary training to ensure familiarity with the associated tasks, considering available job aids, instructions, and mitigating strategy time constraints.

Care has been taken to not give undue weight (in comparison with other training requirements) for Operator training for BDBEE accident mitigation. The testing/evaluation of Operator knowledge and skills in this area has been similarly weighted.

Where appropriate, integrated FLEX drills will be organized on a team or crew basis and conducted periodically; with all time-sensitive actions to be evaluated over a period of not more than eight years. It is not required to connect/operate permanently installed equipment during these drills.

Conclusions

The NRC staff finds that the licensee has adequately addressed the procedures and training associated with FLEX because the procedures have been issued 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 [Reference 30], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 31], 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 including the portable diesel pumps and generators have also been issued.

The licensee has committed to abide by the EPRI generic resolution described above. The FLEX equipment program ensures the equipment is maintained to the standards of NEI 12-06 [Reference 6] Section 11.5, which endorses the guidance of INPO AP 913, "Equipment Reliability Process," and the EPRI associated bases to define site specific maintenance and testing.

Conclusions

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

3.14 Alternatives to NEI 12-06, Revision 0

Permanent Pre-staging of FLEX Diesel Generators

As previously discussed in Section 3.6.1, during the on-site audit, the NRC staff determined that the FLEX strategy to use permanently pre-staged DGs should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. Duke evaluated the strategy regarding the approach for permanently pre-staging two 480V, 830KW FLEX DGs in a single building that is a seismic Category I structure which meets the plant's design-basis with respect to the extreme external hazards. Duke provided a justification for the proposed alternative in their fourth 6-month update [Reference 14] and a position paper. Duke stated that the FLEX DG connection points are also contained within this structure and that ac power from either FLEX DG can be routed through and attached output breaker to a patch panel via portable cables to a 480V to 6.9kV transformer. Power from the 6.9kV transformer is routed through permanent cabling located within safety-related conduit in the seismic category 1 RAB. The cables are routed to medium voltage termination boxes, and either termination box can be connected via portable cables to a FLEX temporary power unit to power the respective 480V bus and the FLEX MCC-RAB. Therefore, the FLEX DGs, associated output devices, and cabling are protected from all external hazards while remaining physically disconnected from the plant's installed electrical distribution system.

Duke's evaluation considered the potential causes of a failure of both FLEX DGs and determined that such a failure is highly unlikely given the design and layout of the FLEX DG installation and the quality of the installed components. The evaluations included, (a) catastrophic mechanical failure of one FLEX DG potentially impacting the other, (b) inadvertent fire suppression system discharge in the FLEX DG building, and (c) fire in the FLEX DG building affecting both FLEX DGs. Duke evaluated FLEX DG installation against the applicable requirements and concluded that the installation constitutes an acceptable alternative to the NRC endorsed guidance provided in NEI 12-06. The NRC staff found the proposed alternative to be acceptable.

(N+1) FLEX Storage Building

As previously discussed in Section 3.6.1, during the audit process the NRC staff determined that the FSB, which houses the (N+1) FLEX ESW pump and associated duplex strainer, discharge manifold hose, trailer, fittings, and tools does not meet the standards of ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures" and should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. Duke agreed that this configuration is an alternative to NEI 12-06.

Duke stated, in its FIP, that the required FLEX equipment may be unavailable for up to 90 days provided that the site FLEX capability (N) is met. If the site FLEX (N) capability is met but not fully protected for the site's applicable extreme external hazards, then the allowed unavailability is reduced to 45 days. In the event that the FLEX ESW Pump stored in the EDG Building Bay 2A is unavailable, then Duke will take the following actions within 45 days: Restore the availability of two FLEX ESW Pumps to the standard storage configuration. Stage the remaining FLEX ESW Pump (N+1) in the EDG Building Bay 2A.

The NRC staff notes that by letter dated December 10, 2015, NEI submitted guidance document NEI 12-06, Revision 2 (ADAMS Accession No. ML16005A625) to the NRC for review. By letter dated January 22, 2016 (ADAMS Accession No. ML15357A163), the NRC staff endorsed NEI 12-06, Revision 2. NEI 12-06, Revision 2 contains modifcations, which resulted in NRC acceptance of the storage of backup (N+1) equipment in a non-robust storage building.

Duke's proposed alternative complies with NEI 12-06, Revision 2. However, Duke will develop procedures to incorporate the maintenance and testing specifications provided in NEI 12-06, Revision 2, Section 11.5, "Maintenance and Testing," Paragraph 4.

The NRC staff reviewed the licensee's proposal and finds that with the additional methods, described above, used to ensure that the primary (N) set of equipment is available, combined with an allowed unavailability to 45 days if any (N) equipment is not protected for all sites applicable hazards, is an acceptable alternative to NEI-12-06, Revision 0.

(N+1) SFP Spray Flow Pumps

As previously discussed in Section 3.3.4.3, during the audit process the NRC staff determined that the licensee's SFP spray flow strategy which uses only one (N) pump, the EDMP, does not meet the N+1 capability required for the equipment described in Table 3-2 and further described in Table D-3, and should be considered an alternative to the guidance in NEI 12-06 [Reference 6]. The licensee agreed that this configuration is an alternative to the N+1 capability guidance in NEI 12-06. However the NRC staff finds this alternative acceptable because of the following reasons:

- The EDMP is stored in the FSB which is fully protected from all applicable external hazards.
- Spray flow was intended to be a defense-in-depth capability and the licensee has multiple, protected, fully capable strategies to provide SFP makeup as described in Section 3.3.
- As described in the NRC memo dated May 21, 2014 (ADAMS Accessio No. ML14136A126), the Harris SSE bounds the reevaluated seismic hazard (GMRS) in the 1-10 Hz range. Therefore, it is unlikely that a seismic event will cause a failure of the SFP requiring spray makeup.

3.15 Conclusions for Order EA-12-049

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

4 .0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 53], the licensee submitted its OIP for HNP in response to Order EA-12-051. By letter dated July 11, 2013 [Reference 54], the NRC staff sent a Request for Additional Information (RAI) to the licensee. The licensee provided its response by letter dated August 12, 2013 [Reference 55]. By letter dated November 19, 2013 [Reference 56], the NRC staff issued an ISE with RAIs to the licensee.

By letters dated August 26, 2013 [Reference 58], February 27, 2014 [Reference 59], August 22, 2014 [Reference 60], and February 23, 2015 [Reference 61], the licensee submitted status reports for the OIP. The OIP describes the strategies and guidance to be implemented by the licensee for the installation of reliable Spent Fuel Pool Level Instrumentation (SFPLI), which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated July 10, 2015 [Reference 18], Duke reported that it had achieved full compliance with the requirements of Order EA-12-051.

The licensee has installed a SFPLI system designed by AREVA. The NRC staff reviewed the vendor's SFPLI system design specifications, calculations, analyses, test plans, and test reports, and issued an audit report of AREVA on September 15, 2014 [Reference 62].

The NRC staff performed the site audit of HNP to review the implementation of SFP level instrumentation related to Order EA-12-051. The scope of the audit included verification of
(a) site's seismic and environmental conditions enveloped by the equipment qualifications,
(b) equipment installation met the requirements and vendor's recommendations, and
(c) program features met the requirements. By letter dated April 14, 2015 [Reference 17], the NRC issued an audit report on the licensee's progress.

4.1 Levels of Required Monitoring

In its OIP, the licensee stated that at HNP there are three separate SFPs (Pools A, B, and C) on the same elevation that are interconnected by normally removed gates, except for limited periods of maintenance or non-refueling operations. Pools A, B, and C are considered a single SFP with one instrumentation channel installed in each pool. A fourth SFP (Pool D) located at the north end of Fuel Handling Building is unused without any stored spent fuel. The SFP level instrument channels will be installed to monitor the water level on the three active SFPs. The bottom of the SFP is at 246 feet plant elevation. The top of the fuel racks in Pools A and B are approximately 260 feet plant elevation and the top of the fuel racks in Pools C is approximately 260.7 feet plant elevation. Therefore, the highest point of any fuel racks with fuel seated in the SFP is approximately 260.7 feet plant elevation. The bottom of the canals at each gate interconnecting Pools A, B, and C are approximately 260 feet plant elevation. Additionally, the bottom of the transfer canal, between Gates 1 and 5, slopes from 260.5 feet plant elevation in approximately the middle of the canal down to approximately 260 feet plant elevation at Gates 1 and 5. Thus the highest point separating Pool C from Pools A and B is 260.5 feet plant elevation which is below the highest point of any fuel racks at 260.7 feet plant elevation.

The NRC staff noted that Level 1 was set at approximately 284.5 feet plant elevation, Level 2 is approximately 270.7 feet plant elevation, and Level 3 is approximately 260.7 feet plant elevation. The approximate range of the instrumentation system will be from 260.7 feet to 284.5 feet plant elevation.

In its letter dated Aug 12, 2013 [Reference 55], the licensee provided the revised Level 1 and the basis for the change: Level 1 is set at 282 feet elevation to assure that the pool suction line inlets are covered. This is based on the following:

- The SFP cooling pump test report indicates that required net positive suction head (NPSH) is 18 feet. The SFP cooling pump total discharge head calculation assumes that NPSH is 19 feet. A NPSH of 19 feet corresponds to 257.8 feet elevation.
- The pool suction line inlets are located at 277.5 feet elevation for pool "A" and at 278.5 feet elevation for pools "B" and "C."
- 282 feet elevation corresponds to the current SFP lo-lo level alarm and provides margin above the pool suction line inlets and NPSH.

The NRC staff noted that Level 1 designation is adequate for normal SFP cooling system operation and it is also adequate to ensure the required SFP cooling pump NPSH. Level 2 designation was determined using the first of the two options described in NEI 12-02 [Reference 8] for Level 2, which is approximately 10 ft. above the top of the fuel rack. Level 3 designation is the top of the fuel rack.

However, the NRC staff questioned how the instruments would respond under various combinations of gate closures. In response to the NRC staff's concern, in its compliance letter [Reference 18], the licensee stated that Figure 14A shows the approximate location of each channel and the associated power supply for the SFPLI that monitors the SFPs.

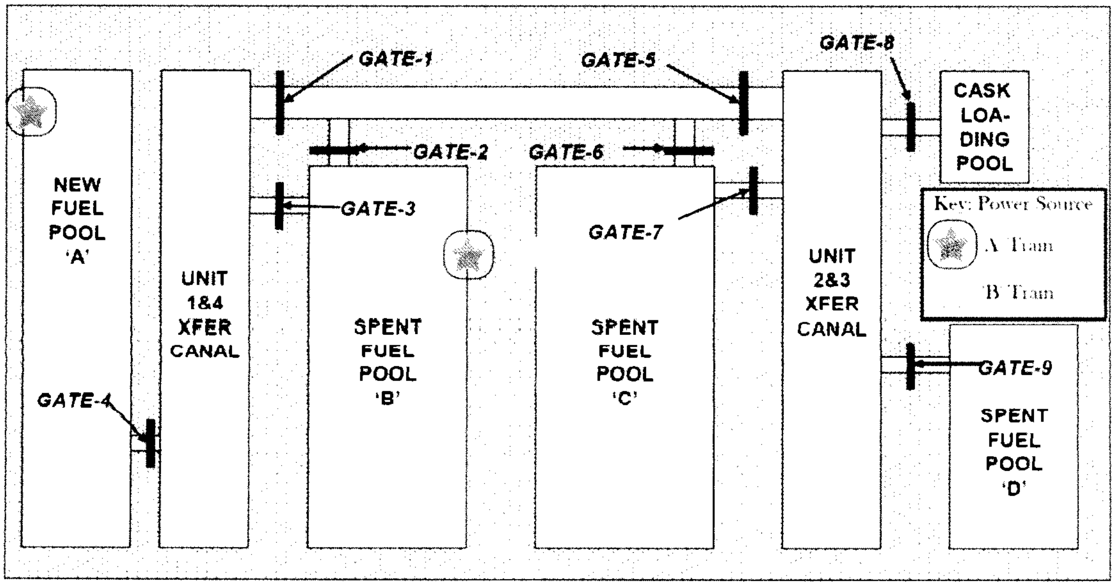

Figure 14A

Based upon the pool interconnections and gate locations shown in Figure 14A, there are numerous gate installation combinations that may impact availability of the SFP level instrument channels. Examples of various gate combinations and the impact upon instrument availability are provided in the table below. Examples 4 and 5 also incorporate the combined effect of gate isolation and requirements for independent power sources. The impact of gate configuration, power isolation, and functionality is described below the table to support the examples identified.

Example #	Gates Installed and Inflated	Availability Impact
1	Gate 4	A Pool: only 1 channel available
2	Gates 3 and 1	A Pool: only 1 channel available
3	Gates 3 and 2	B Pool: only 1 channel available
4	Gates 6 and 7	A, B, C Pool: only 1 channel available (each)
5	Gates 6 and 5	A, B, C Pool: only 1 channel available (each)

Gate Configuration:

A chanel installed in one pool is considered to be available to monitor level for another pool as long as gates have not isolated the pools from each other (e.g., there is physical connection of the pools).

Independent Power Sources:

Primary and back-up instrument channels must have independent power supplies. For a pool to have both primary and back-up channels available, one must be powered from "A" Train and the other from "B" Train. Backup battery does not count as an independent power supply.

Functionality:

- Pool level indication is available in the MCR Complex and has satisfactorily passed required channel-checks.
- Satisfactory performance of PIC-I709 per PMR 714847 that includes the replacement of backup batteries (within 60 days prior to each 18 months scheduled refueling outage).

The NRC staff found the licensee adequately addressed the required monitoring levels including possible pool interconnections and separation conditions with gate combinations, and the available number(s) of level indication for each pool under those conditions. Subsection 4.3.3, "Programatic Controls: Testing and Calibration," further addresses compensatory measures to assure that reliable level monitoring is available for each pool under out-of-service conditions of instrument channels.

Based on the discussion above, the NRC staff finds that the licensee's proposed Levels 1, 2, and 3, and the number of instruments to monitor the pools, are 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. The specific requirements are outlined in Section 2.2 of this safety evaluation and are evaluated below.

4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the instrumentation will consist of a single separate permanent fixed instrument channel per pool (total 3 channels) to monitor SFP water level continuously, from normal water level (approximately 284.5 feet plant elevation) down to a level at the highest point of any fuel racks at approximately 260.7 feet plant elevation.

The NRC staff noted that the range specified for the licensee's instrumentation will cover Levels 1, 2, and 3 as described in Section 4.1 above. The NRC staff finds that the licensee's design, with respect to the number of channels measurement range for its SFP, is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.2 Design Features: Arrangement

In its OIP, the licensee stated that the level instrument channels will be installed in diverse locations and physically arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the SFP. The sensing component of each level instrument channel will be installed separately within the SFP in order to reduce common susceptibility to missiles and other external events. Electronics associated with the level instrumentation will be located outside the SFP operating area due to sensitivity to radiation. Cable routings will be installed that will provide reasonable protection from missiles that may result from damage to the structure over the SFP and refuel floor. The conduit and cable routing will be determined by the detailed design.

In its compliance letter dated July 10, 2015 [Reference 18], the licensee provided locations of the SFP level sensors and the routing of the cables in the following drawings:

- Figure 1A, " EC 89579 Fuel Pool Level Indication Schematic"
- Figure 1B (SK-94499-C-1000), "Plan View-Spent Fuel Pool 'A'"
- Figure 1C (SK-94499-C-1000), "Elev. view-Spent Fuel Pool 'A' Looking West"
- Figure 1D (SK-94499-C-1000), "Plan View-Spent Fuel Pool 'B'"
- Figure 1E (SK-94499-C-1000), "Elev. View-Spent Fuel Pool 'B'"
- Figure 1F (SK-94499-C-1000), "Plan View-Spent Fuel Pool 'C'"
- Figure 1G (SK-94499-C-1000), "Elev. view-Spent Fuel Pool 'C' Looking West"
- Figure 1H, "General Arrangement Fuel Handling Building Plant – Sheet 1"

The NRC staff verified by walkdown during the onsite audit, that there is sufficient channel separation within the SFP area between the primary and back-up level instrument channels, sensor electronics, and routing cables to provide reasonable protection against loss of indication of SFP level due to missiles that may result from damage to the structure over the SFP.

The NRC staff finds that the licensee's arrangement for the SFP level instrumentation, if implemented appropriately, is 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 Design Features: Mounting

In its OIP, the licensee stated that each permanently installed instrument channel will be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the SFP structure.

In its letter dated July 10, 2015 [Reference 18], the licensee further stated that all equipment installed is qualified in accordance with IEEE Standard 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations". The equipment installed at the SFP edge is qualified to withstand the maximum hydrodynamic forces from seismically induced wave action (sloshing). The HNP SFPLI utilizes a waveguide pipe with horn end assembly, control panel, sensor, display and mounting brackets. This system has been seismically tested to the requirements of IEEE 344-2004. The required response spectra (RRS) used from the shake table test has a peak horizontal and vertical acceleration 14g for SSE and approximately 10g for the operating basis earthquake (OBE). This RRS bounds the HNP response spectra for the Fuel Handling Building. The horn end assembly is qualified for a vertical and side pressure (slosh) loading of 3.37 psi. Actual loading from impact and drag due to slosh is calculated to be a maximum vertical pressure (slosh) loading of 1.09 psig and a maximum side pressure (slosh) loading of 0.505 psig. Therefore, the horn end assembly and horn cover are structurally qualified for seismic and hydrodynamic loading including the horn cover. The mounting design for the power control panel is qualified considering the total weight of the panel, its associated components, and the seismic accelerations for the building structure. All of the mounting supports for the waveguide piping are attached to either the concrete wall or concrete floor. These concrete structures have a minimum concrete strength of 4000 psi.

The affected structures and equipment were qualified using American Institute of Steel Construction (AISC) Code 8th, 9th, and 13th Edition for structural steel elements and ASME B31.1-1986 Edition Anchorage Code for Pressure Piping. The Fuel Handling Building design specifications (CAR1364.481S04) provide the design criteria and loads (including allowances) for electrical and mechanical components.

Waveguide pipe is mounted in the FHB at the 286' and 261' elevations. Supports consist of seismically prequalified HNP support designs in accordance with the HNP generic piping support drawings 5-G-0107 Sheet 3 and 5-G-0107 Sheet 4 and engineered seismically qualified supports. Interim supports between horn end and sensor are seismically qualified in the Spent Fuel Pool Wave Guide Support Qualification document. The interim supports are designed to HNP's peak DBE seismic accelerations using 2 percent damping. The channel mounted horn end assembly and sensor supports are qualified by AREVA and are calculated in Document 32-9221237-003, "Qualification for a Waveguide Type "A" Support and Horn Assembly for AREVA Spent Fuel Pool Level Monitoring Instrumentation".

During onsite audit, the NRC staff verified the instrument channel's mounting design by performing a walkdown and by reviewing the following documents and drawings.

- Engineering Change (EC) 89579, "Fukushima Response Project – SFP Wide Range Level Indication – HNP," Revision 0 [Reference 80]
- Sloshing Analysis HNP-C/FLEX-0003, "Seismic Induced Hydraulic Response in the Harris Spent Fuel Pool," Revision 0 [Reference 81]
- Calculation 32-9221971-000, "Qualification of Difference in as-tested and as-fabricated Mounting Configuration for VEGA Power Control Panel Mounting Assembly," Revision 0 [Reference 82]
- Drawing SK-89579-C-1001, Sheet 1 thru 4, "SFPLI Wave Guide Pipe Supports," Revision 0 [Reference 83]
- Drawing SK-89579-C-1002, Sheet 1 thru 6, "Power Control Panel Assembly Mounting Locations, Sections, and Details," Revision A [Reference 84]
- Drawing SK-89579-C-1003 (Areva Drawing 02-9214553B), "Horn and Sensor Support SFP A," Revision 0 [Reference 85]
- Drawing SK-89579-C-1004 (Areva Drawing 02-9214554B), "Horn and Sensor Support SFP B," Revision 0 [Reference 86]
- Drawing SK-89579-C-1005(Areva Drawing 02-9214555B), "Horn and Sensor Support SFP C," Revision 0 [Reference 87]
- Drawing Markup CAR-2165 G-022, "General Arrangement Fuel Handling Building Plants – Sheet 1," Revision 0 [Reference 88]

Based on its review of the above documents, the NRC staff found that the licensee adequately addressed the SFPLI mounting requirements. The methodology, design criteria, assumptions, and analytical model used in the used to estimate and test the total loading on the mounting devices, including the design-basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing, are adequate. The site-specific seismic analyses demonstrated that the SFPLI's mounting design is satisfactory to allow the instrument to function in accordance with the design following the maximum seismic ground motion. However, the NRC staff noted that the sloshing analysis did not address the hydrodynamic impact for all three pools where the SFP level sensors located. Pools 'A' and 'B' were not analyzed for the hydrodynamic impact. In response to the NRC staff's concern, the licensee stated that in accordance with Calculation NAI-1725-003, "GOTHIC Verification and Sensitivity Studies for Predicting Hydrodynamic Response to Acceleration in Rectangular Shaped Pools," Revision 0 [Reference 89] damping of the slosh height associated with the convective response is highest for pools of short length and the damping tends toward zero as the width of the pool increases. From this, it is evident that modeling the seismic event for the long axis of a rectangular pool is conservative as it lessens damping of the convective response. Hydrodynamic response of liquid filled pools is highly sensitive to the pool size and shape. Of the three SFPs A, B and C, SFPs B and C are same width and length, 27' x 50'. SFP A is smaller in size, 13' x 38'. Based on the sensitivity study and the above discussion, SFP C was utilized in the sloshing analysis, HNP-C/FLEX-0003, to determine the hydrodynamic impact and drag forces on the SFPLI waveguide horn assembly. The licensee stated that this pool selection gives more conservative results.

The NRC staff found that the licensee adequately addressed the NRC staff's concern and verified the response by reviewing Calculation NAI-1725-003 [Reference 89]. Based on the discussion above, the NRC staff finds 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 [Reference 8] describes a quality assurance process for non-safety systems and equipment that is not already covered by existing quality assurance requirements. As discussed 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 Augmented Quality provisions will be applied to ensure that rigor of the qualification documentation reviews and in-plant modification installation oversight is sufficient to ensure compliance with the qualification requirements.

The NRC staff noted that the licensee's proposed augmented quality assurance process appears to be consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03. If implemented appropriately, this approach is consistent with NEI 12-02 guidance, and should adequately address the requirements of the order.

4.2.4.2 Equipment Reliability

The NRC staff reviewed the AREVA SFPLI qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic [Reference 62]. The NRC staff further reviewed the anticipated Harris' environmental conditions during the on-site audit [Reference 17].

4.2.4.2.1 Temperature, Humidity, and Radiation

For the instrument channel's temperature and humidity qualifications and associated Harris' site conditions, in its letter dated July 10, 2015 [Reference 18], the licensee stated that the postulated temperature and humidity in the SFP room that results from a boiling pool is 100 degrees Celsius (°C) (212°F) with saturated steam. The electronics in the sensor are rated for a maximum continuous duty temperature of 80°C (176°F) on the condition that the process temperature (that which the flange connection is in contact with) is no greater than 130°C (266°F). The sensor must be located away from the SFP in an area where the temperature is at or below the rated temperature. The sensor has been tested in accordance with IEC 60068-2-30, which varies the temperature from room temperature to elevated temperature at high humidity conditions to verify that the test item withstands condensation that can occur due to the changing conditions. The power control panel internal components are rated for a maximum temperature of at least 70°C (158°F). Allowing for 5°C (9°F) heat rise in the panel, the overall panel maximum ambient temperature for operation is 65°C (149°F). The power control panel enclosure is rated NEMA 4X and provides protection to the internal components from the effects of high humidity environments. Condensation formation on the inner waveguide pipe walls would require very moist air to enter the pipe at the sensor and travel to a colder area where the air temperature in the pipe would be lowered to the dew point. This is a highly unlikely occurrence given the limited length of waveguide. The horn cover, which blocks airflow through the waveguide pipe, reduces the potential for transfer of warm moist air to a colder area and therefore reduces the potential for condensation forming in the pipe. Normal operating conditions for the control panel, transmitter, and display for each SFPLI unit are considered to be a mild environment.

For the instrument channel's radiation qualifications and Harris' radiological conditions, in its letter dated July 10, 2015 [Reference 18], the licensee stated that the area above and around the pool will be subject to large amounts of radiation in the event that the fuel becomes uncovered. The only parts of the measurement channel in the pool radiation environment are the metallic waveguide, horn, and fused silica glass horn cover, which are not susceptible to the expected levels of radiation. The sensor electronics are located within the Fuel Handling and Waste Processing Buildings at an elevation below the SFP operating deck. The location of the electronics is shielded from the direct shine from the fuel, to result in bounce and scatter effects above the pool. The sensor is shielded from SFP area dose during a postulated event with SFP water level at level 3 by a 6 foot thick reinforced concrete wall including a 3/16" steel liner plate between the SFP and the Fuel Handling and Waste Processing Buildings and by the 2 foot thick reinforced concrete floor below. Based on vendor calculations, the maximum calculated dose over the 168 hour time period the sensor would receive is approximately 0.4 rads. Therefore, during a postulated BDBEE, the sensor total integrated dose is enveloped by the vendor instrumentation design limit of 1×10^3 rads.

During the onsite audit, the NRC staff reviewed EC 91690 [Reference 77] to verify that the equipment qualifications bound the temperature conditions at locations where the electronics located. According to the analysis in EC 91690:

- Maximum temperature at I&C Shop (261-foot elevation), where Channel 'A' power control panel, indicator, and transmitter located, is 123°F;
- Maximum temperature at SFP A&B Demineralizer (261-foot elevation), where Channel 'B' power control panel, indicator, and transmitter located, is 139°F;
- Maximum temperature at SFP C&D Demineralizer (261-foot elevation), where Channel 'C' power control panel, indicator, and transmitter located, is 139°F; and
- Channel "B" and "C" SFP level instrumentation is located on the 261' elevation of the Fuel Handling Building. Channel "A" SFP level instrumentation is located in the I&C shop on the 261' elevation in the Waste Processing Building. The SFP level instrumentation locations are considered mild since these areas are not EQ Zones.

The NRC staff found that the licensee adequately addressed the equipment reliability of SFPLI with respect to temperature, humidity and radiation. The equipment qualification envelops the expected Harris' radiation, temperature, and humidity conditions during BDB event. The equipment environmental testing demonstrated that the SFP instrumentation will maintain its functionality during the expected BDB conditions.

4.2.4.2.2 Shock and Vibration

For the instrument channel's shock and vibration qualifications, in its letter dated July 10, 2015 [Reference 18], the licensee stated that the through air radar sensor was shock tested in accordance with MIL-S-901D, and vibration tested in accordance with MIL STD 167-1. The MIL-S-901D test consisted of a total of nine (9) shock blows, three (3) through each of the three (3) principal axes of the sensor, delivered to the anvil plate of the shock machine. The heights of hammer drop for the shock blows in each axis were one foot, three feet and five feet. The MIL STD 167-1 vibration test frequencies ranged from 4 Hz to 50 Hz with amplitudes ranging from 0.048 inches at the low frequencies to 0.006 inches at the higher frequencies. The potential vibration environment around the SFP and surrounding building structure might contain higher frequencies than were achieved in the testing discussed above. However, in addition to the MIL Standard testing above, the sensor has been shock tested in accordance with EN 60068-2-27, (100 g, 6 ms), and vibration tested in accordance with EN 60068-2-6, Method 204 (except 4g, 200 Hz). The power control panel is shock tested in accordance with EN 60068-2-27 and vibration tested in accordance with EN 60068-2-6.

The NRC staff found the licensee adequately addressed the equipment reliability of SFPLI with respect to shock and vibration qualifications. The NRC staff also reviewed the shock and vibration test report during the vendor audit and found it acceptable.

4.2.4.2.3 Seismic

For the instrument channel's seismic qualifications and Harris' response spectra, in its letter dated July 10, 2015 [Reference 18], the licensee stated that the SFPLI system has been seismically tested to the requirements of IEEE 344-2004 and the equipment is considered operable for accident conditions, including a seismic event. The equipment qualified included the VEGAPULS 62 ER sensor, PLICSCOM indicating and adjustment module, VEGADIS 62 display, power control panel, rotatable horn waveguide assembly, waveguide piping including standard and repair flanges, and pool end and sensor end mounting brackets. The RRS used from the shake table test has a peek horizontal and vertical acceleration 14g for SSE and approximately 10g for OBE. This RRS bounds HNP response spectra for the Fuel Handling Building. The horn end assembly is qualified for a vertical and side pressure (slosh) loading of 3.37 psi. Actual loading from impact and drag due to slosh is calculated to be a maximum vertical pressure (slosh) loading of 1.09 psig and a maximum side pressure (slosh) loading of 0.505 psig. Therefore the horn end assembly is structurally qualified for seismic and hydrodynamic loading including horn cover.

The NRC staff found the licensee adequately addressed the equipment reliability of SFP level instrumentation with respect to seismic qualification. Further discussion of the instrument channel's mounting design is described in Subsection 4.2.3, "Design Features: Mounting". The equipment's seismic qualifications envelop the expected Harris' seismic condition during

BDBEE. The site-specific seismic analyses demonstrated that the SFPLI's mounting design is satisfactory to allow the instrument to function per design following the maximum seismic ground motion. However, during the onsite audit the NRC staff learned that Duke installed the Weschler, Model VX-252, analog indicators in the MCR for the credited SFP level indication, and no qualification information for these indicators available for review. In response to the NRC staff's concerns, the licensee stated that Report ER-20130518-01, "Report of Qualification Testing" for indicator model VX-252 indicates that the indicator was tested at 25°C ± 10°C and 90 percent relative humidity. The NRC staff found the licensee's response adequately addressed the staff's concern as the Weschler indicators are qualified for the control room environment. In addition, the NRC staff reviewed the vendor's factory acceptance test reports and found the Harris SFPI design and qualification process acceptable. The environmental conditions at the Harris' SFPLI equipment locations enveloped by the vendor's equipment qualification.

Based on the discussion above, the NRC staff finds 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 <u>Design Features: Independence</u>

In its OIP, the licensee stated that the instrument channels for the SFPs will be separated by distance and electrically independent of one another. The channels will have their own sensing components reasonably separated, separate cable routes, and separate electronics.

In its letter dated July 10, 2015 [Reference 18], the licensee further stated that RAI-1 Response, Figure 1A provides a schematic representation of the physical and spatial separation of three SFPLI channels including the sensors, transmitter and control panels. Instrument for SFPLI channels "A" and "B" are powered by the same source, (MCC 1&4A33-SA, power panel PP-1&4A33 breakers 40 & 42), however SFLPI "C" is powered by an independent power source (MCC 1&4B33-SB, power panel PP-1&4B33 breaker 38). Each SFLPI has independent replaceable batteries with sufficient capacity to maintain the level indication function until the normal power is restored. The NRC staff verified the instrument channels' physical and electrical independence by the walkdown and by reviewing the following:

- Figure 1A, "Fuel Pool Level Indication Schematic," of EC 89579, ""Fukushima Response Project – SFP Wide Range Level Indication – HNP"

- Drawing CAR 2166-B-041 Sheet 650, " Unit No. 1 Power Distribution & Motor Data 208/120V Power Panel PP-1&4A33-SA"

- Drawing CAR 2166-B-041 Sheet 651, " Unit No. 1 Power Distribution & Motor Data 208/120V Power Panel PP-1&4B33-SB

The NRC staff noted that the licensee adequately addressed the instrument channel independence. The primary instrument channel is physically and electrically independent of the backup instrument channel. The instrument channels' physical separation is discussed in Subsection 4.2.2, "Design Features: Arrangement". With the licensee's proposed power arrangement, the electrical functional performance of each level measurement channel would be considered independent of the other channel, and the loss of one power supply would not affect the operation of other independent channel under BDBEE conditions.

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

4.2.6 Design Features: Power Supplies

In its letter dated July 10, 2015 [Reference 18], the licensee stated that vendor analyses support the battery capacity (at 20 milliamp (mA) continuous discharge) as presented in table 1 below. The calculated battery backup times below demonstrate that the backup battery has sufficient capacity to support reliable instrument channel operation until offsite resources can be deployed by the mitigating strategies in response to Order EA-12-049.

Table 1: Backup Battery Lifetimes vs. Temperature

Temperature (Degrees)	Lifetime at full voltage (Hours) @ 20 mA (hours)
-30°C (-22°F)	131
0°C (32°F)	233
25°C (77°F)	330
55°C (131°F)	349
75°C (167°F)	209

The selection of the ac power sources is discussed in Subsection 4.2.5, "Design Features: Independence".

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

4.2.7 Design Features: Accuracy

In its letter dated July 10, 2015 [Reference 18], the licensee stated that the vendor factory acceptance test demonstrated reliable operation of the SFPLI under normal conditions and under various simulated test conditions (e.g. steam exposure). The testing demonstrated the instrumentation met design accuracy and repeatability specifications. Under normal conditions, the instrument accuracy is ±1 inch, and error due to all effects including 212°F saturated steam is ±3 inches. The instrument accuracy was verified during factory acceptance testing.

The NRC staff reviewed the AREVA SFP instrument test report for the accuracy during the vendor audit [Reference 62] and found the accuracy to be acceptable. During the onsite audit, the NRC staff asked for information related to the accuracy of the Weschler, Model VX-252 indicator. In response, the licensee provided Report ER-20130518-01, "Report of Qualification

Testing" (for Weschler indicator model VX-252), which indicates that the indicator was tested with the accuracy acceptance criteria of ±1.5 percent. The NRC staff found that the accuracy for the Weschler indicators acceptable.

Based on the discussion above, the NRC staff finds 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 letter dated July 10, 2015 [Reference 18], the licensee stated that periodic testing will require taking measurements for validating accuracy and performing any adjustments to bring accuracy to within the required tolerances. Calibration Procedure PIC-I709 is used for the calibration of AREVA SFPLI Radar Measurement System. The SFPLI scaling is from the top of any fuel racks (at approximately 261' plant elevation pool level) up to SFP level at the 285' elevation. The SFP level range meets the requirements of NRC Order EA-12-051. A calibration tolerance of ±1 inch was used as the acceptance criteria for the factory acceptance testing. The conditions for the HNP SFP area are different from the laboratory conditions present for the vendor testing. The system target and horn section are located on the 286' elevation where verification of the system reading will be made and the system itself is located on the 261' elevation. At HNP, all SFPs are interconnected and when separated by removal gates, each channel is independent. The indication of one independent SFP channel can be compared against another pool channel, in addition to the fixed level indication marks on the SFP wall.

Calibration Procedure PIC-I709 provides the instructions for testing and calibrating the SFPLI equipment. This procedure performs removal from service, calibration check, calibration, return to service, and post maintenance activities. Monthly channel checks of the SFP level indication will be performed on AEP-2 in Modes 1, 2, and 3 using plant procedure OST-1020, "Remote Shutdown Monitoring and Accident Monitoring Instrumentation Channel Check Monthly Interval Modes 1-2-3," to ensure the channel A, channel B, and channel C remote indications are aligned. Weekly channel checks will be performed on AEP-2 in Modes 3 and 4 per OST-1022, "Daily Surveillance Requirements Daily Interval Modes 3 and 4," and in Modes 5, 6, and Defueled per OST-1033, "Daily Surveillance Requirements Daily Interval Mode 5, 6, and Defueled," to ensure the channel A, channel B, and channel C remote indications are aligned. If a gate configuration is present that isolates one pool from the other connected pools, pool level will be verified locally and compared to the level indication on AEP-2.

A preventative maintenance (PM) activity will allow for the calibration of the SFPLI equipment on an 18-month basis, prior to scheduled refueling cycles. This PM activity will validate the level instrumentation values against the SFP ruler and against each other to ensure functionality of the system. This testing includes the performance of a channel check and functional verification of the primary channel battery back-up capability. Procedure PIC-I709 will be performed during this PM activity in accordance with the PM requirements.

The NRC staff found the responses adequately addressed periodic testing and calibration and preventive maintenances. During the onsite audit, the NRC staff verified the responses by reviewing PIC-I709, "Radar Wave Guide System Spent Fuel Pool Level Instrumentation," Rev. 0, and CM-I0084, "Methodology of Scale Replacement for VX-252 Indicators," Rev. 0. The

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

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

4.2.9 Design Features: Display

In its letter dated July 10, 2015 [Reference 18], the licensee stated that the remote indicator for each of the three SFPs are located on AEP-2 in the Control Room back panel area. Local indicators are mounted at the power control panel (near the instrumentation that is located on the 261' elevation of the Fuel Handling and Waste Processing Buildings). Results from Calculation HNP-M/FLEX-0013, "Harris Fuel Handling Building Extended Loss of AC Power FLEX Response," Revision 0, [Reference 78] show the expected maximum temperature in this area of approximately 140°F. The time it takes for an operator to travel from the Control Room to each indicator display is less than 5 minutes, based upon walk-through.

The NRC staff noted that the licensee adequately addressed the display requirements. If implemented properly, the displays will provide continuous indication of SFP water level. The displays are located in seismically qualified buildings and the accessibility of the MCR and Waste Processing Building following an ELAP event is considered acceptable.

The NRC staff finds that the licensee's proposed location and design of the SFP instrumentation displays 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 Evaluation of Programmatic Controls

Order EA-12-051 specified that the SFPLI 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 SFPLI.

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that the SAT will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training. Training will be completed prior to placing the instrumentation in service.

The NRC staff finds that the use of SAT to identify the training population and to determine both the elements of the required training is acceptable. The licensee's proposed plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFPLI and the provision of alternate power to the primary and backup instrument channels, including the approach to identify the population to be trained, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.2 Programmatic Controls: Procedures

In its OIP, the licensee stated that procedures will address strategy to ensure SFP water addition is initiated at an appropriate time consistent with implementation of NEI 12-06 [Reference 6].

In its letter dated July 10, 2015 [Reference 18], the licensee provided a list of procedures addressing operation, calibration, test, maintenance, and inspection procedures that have been developed for the SFPLI as below.

Procedure Type	Title	Number	Objective
Abnormal Response	Spent Fuel Pool Events	AOP-041	This procedure provides instructions for responding to events that result in a loss of inventory or increased temperature in the fuel pools containing spent fuel or the SFP Cooling System.
Calibration/ testing	Radar/Waveguide System Spent Fuel Pool Level Instrumentation	PIC-I709	This procedure contains the instructions for calibration of the level indicators, level sensors, controllers, and transmitters (preventative maintenance).
Maintenance	Methodology of Scale Replacement for VX-252 Indicators	CM-I0084	This procedure establishes a standard method of electrical testing before scale replacement, during scale replacement, and electrical testing after scale replacement for model VX-252 indicators.
Maintenance	Preventative Maintenance work order instruction	PMR 714847	The preventative maintenance work activity to complete PIC-I709 will include a step to check the PCP battery component.
Operations	Remote Shutdown Monitoring and Accident Monitoring Instrumentation Channel Check Monthly Interval Modes 1-2-3	OST-1020	This procedure contains a step for completing channel checks of the SFPLI on a monthly frequency in Modes 1, 2, and 3.
Operations	Daily Surveillance Requirements Daily Interval Modes 3 and 4	OST-1022	This procedure contains a step for completing channel checks of the SFPLI on a weekly frequency in Modes 3 and 4.
Operations	Daily Surveillance Requirements Daily	OST-1033	This procedure contains a step for completing channel checks of the SFPLI on a weekly frequency in Modes 5, 6, and Defueled.

	Interval Mode 5, 6, and Defueled		
Administrative	FLEX Strategies And Equipment Availability	PLP-137	This procedure establishes programmatic controls for FLEX equipment to provide reasonable assurance that equipment is maintained available to implement FLEX strategies required to mitigate a Beyond Design-Basis event.

The NRC staff noted that the licensee adequately addressed the procedure requirements. The procedures had been established for the testing, surveillance, calibration, and operation of the primary and backup SFPLI channels. The NRC staff finds that the licensee's proposed procedures appear to be consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

In its OIP, the licensee stated that testing and calibration of the instrumentation will be consistent with vendor recommendations or other documented basis. Calibration will be specific to the mounted instruments and the displays. A maintenance procedure will be written to direct calibration and repair of the instruments.

In its letter dated July 10, 2015 [Reference 18], the licensee stated that the preventative maintenance activity that allows for the performance of periodic instrument calibration verification, functional tests, and maintenance for the instrument channels is implemented through the HNP Preventative Maintenance Program. Periodic channel checks are completed by Operations, and the channel out of service durations, required remedial actions, and required action timeframes are formally monitored by Operations, whom refer to instructions contained in the Administrative Procedure, PLP-137, "FLEX Strategies And Equipment Availability." PLP-137 contains the compensatory measures for a condition in which one or both SFPLI channels are determined to be non-functional. Compensatory actions for a single instrument channel out of service or unavailable (e.q. maintenance or SFP gate(s) closed) beyond 90 days could include one or more of the following actions, which would be implemented within 72 hours:

a. Increased surveillance (channel check) to verify functionality of the remaining level channel
b. Implementation of equipment protection measures
c. Increased operator visual surveillance of the SFP level and area
d. Maintain elevated SFP level
e. Reduce SFP temperature
f. Supplemental Operations staffing

Compensatory actions for both the primary and back-up level channels out of service could include one or more of the following, which would be implemented within 72 hours:

a. Increased operator visual surveillance of the SFP level and area

b. Maintain elevated SFP level
c. Reduce SFP temperature
d. Supplemental Operations staffing
e. Pre-stage FLEX support equipment (nozzles, hoses, etc.) which are relied upon for SFP make-up

In its letter dated July 10, 2015 [Reference 18], the licensee further stated that during the 90-day time period established for completion of maintenance activities to restore instrument functionality, SFP level is monitored by the remaining instrument(s) that are available and channel checks are completed at a monthly frequency or a weekly frequency, dependent upon the plant operating mode. Any activities requiring pool isolation will have to account for the unavailable channel so not to render any pool without one instrument channel available or a 24-hour action time will need to restore the channel of instrumentation and to implement compensatory actions within 72 hours.

The NRC staff noted that the licensee adequately addressed necessary testing and calibration of the primary and backup SFPLI channels to maintain the instrument channels at the design accuracy. The testing and calibration are consistent with the vendor recommendations. Additionally, compensatory actions for instrument channel(s) out-of-service appear to be consistent with guidance in NEI 12-02 [Reference 8].

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

4.4 Conclusions for Order EA-12-051

In its letter dated July 10, 2015 [Reference 18], the licensee stated that they would meet the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee has conformed to the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In addition, the NRC staff concludes that if the SFPLI is installed at HNP 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 auditing the licensee's progress on Orders EA-12-049 and EA-12-051. The NRC staff conducted its onsite audit in December 2014 [Reference 17]. The licensee reached its final compliance date on May 14, 2015, and has declared that HNP is in compliance with the orders. The purpose of this SE is to document the strategies and implementation features that the licensee is using to comply with the orders. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs that, if implemented appropriately, will 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 is in compliance with the orders.

6.0 REFERENCES

1. SECY-11-0093, "Recommendations for Enhancing Reactor Safety in the 21st Century, the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated July 12, 2011 (ADAMS Accession No. ML11186A950)

2. SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A103)

3. SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 - Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012 (ADAMS Accession No. ML120690347)

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

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

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

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

8. NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1, dated August 24, 2012 (ADAMS Accession No. ML12240A307)

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

10. Shearon Harris, Unit 1 "Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 28, 2013 (ADAMS Accession No. ML13112A020)

11. Sharon Harris Nuclear Power Plant, "First Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for

Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated August 27, 2013 (ADAMS Accession No. ML13239A359)

12. Shearon Harris, Unit 1, "Second Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 27, 2014 (ADAMS Accession No. ML14072A051)

13. Shearon Harris, Unit 1, "Third Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated August 25, 2014 (ADAMS Accession No. ML14241A115)

14. Shearon Harris, Unit 1, "Fourth Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 23, 2015 (ADAMS Accession No. ML15055A101)

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

16. Letter from Jeremy S. Bowen (NRC) to Ernest J. Kapopoulos, Jr. (Progress Energy) regarding Shearon Harris Nuclear Power Plant, Unit 1 – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigating Strategies) (TAC NO. MF0874), dated February 12, 2014 (ADAMS Accession No. ML13364A214)

17. Letter from Stephen Monarque (NRC) to Benjamin C. Waldrep (Duke Energy) regarding Shearon Harris Nuclear Power Plant, Unit 1 – Report for the Onsite Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Instrumentation Related to Orders EA-12-049 and EA-12-051, dated April 14, 2015 (ADAMS Accession No. ML15083A024)

18. Shearon Harris, Unit 1 - Notification of Full Compliance with Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" and with Order EA-12-051, "Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation," dated July 10, 2015 (ADAMS Accession No. ML15192A006) (nonpublic)

19. NRC, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012, (ADAMS Accession No. ML12053A340)

20. SRM-COMSECY-14-0037, "Staff Requirements – COMSECY-14-0037 – Integration of Mitigating Strategies For Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards," dated March 30, 2015, (ADAMS Accession No. ML15089A236)

21. Shearon Harris, Unit 1, Flooding Hazard Reevaluation Report, dated March 12, 2013 (ADAMS Accession No. ML13079A253)

22. Shearon Harris Nuclear Power Plant, Unit 1, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1, 2.3 and 9.3 of the Near-Term Task Force Review, dated March 27, 2014 (ADAMS Accession No. ML14090A441)

23. EPRI Report 1025287, Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (ADAMS Accession No. ML12333A170)

24. EPRI Final Draft Report, 3002000704, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Near-Term Task Force Recommendation 2.1: Seismic (ADAMS Accession No. ML13102A142)

25. NRC endorsement of Augmented Approach letter, dated May 7, 2013 (ADAMS Accession No. ML13114A949)

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

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

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

29. PWROG-14015, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power," Revision 0

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

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

32. Letter from David L. Skeen (NRC) to Joseph E. Pollock (NEI), regarding "Trigger Conditions for Performing an Integrated Assessment and Due Date for Response", dated December 3, 2012 (ADAMS Accession No. ML12326A912)

33. Letter from William M. Dean (NRC) to all Power Reactor Licensees regarding "Response Requirements For Request For Information Pursuant To Title 10 Of The Code Of Federal Regulations 50.54(f) Regarding Flood Hazard Integrated Assessments for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Khi Accident", dated May 26, 2015 (ADAMS Accession No. ML15112A051)

34. Letter from Andrew Hon (NRC) to Ernest J. Kapopoulos (Duke) regarding "Shearon Harris Nuclear Power Plant, Unit 1, Request for Additional Information Regarding Fukushima Lessons Learned – Flooding Hazards Reevaluation Report", dated February 10, 2014 (ADAMS Accession No. ML14030A419)

35. Letter from Duke Energy Progress, Inc. to NRC, regarding Flood Hazard Reevaluation Report, Revision 1, dated April 1, 2015 (ADAMS Accession No. ML15091A590)

36. Letter from Juan F. Uribe (NRC) to Benjamin C. Waldrep (Duke) regarding "Shearon Harris Nuclear Power Plant, Unit 1, - Staff Assessment of Response to 10 CFR 50.54(f) Information Request Flood Causing Mechanism Reevaluation (TAC No. MF1103), dated April 29, 2015 (ADAMS Accession No. ML15104A370)

37. JLD-ISG-2012-06, NRC "Guidance for Performing a Tsunami, Surge, or Seiche Hazard Assessment," Interim Staff Guidance, dated January 4, 2013, Revision 0 (ADAMS Accession No. ML12314A412)

38. PWROG-14027, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power, Task 3: Evaluations of Revised Seal Flow Rate on Time to Enter Reflux Cooling and Time at which the Core Uncovers," Revision 3

39. CN-CCOE-14-2, "Supporting Chemistry Calculations for Alternate Cooling Source Usage During Extended Loss of All A.C. Power at Shearon Harris Nuclear Power Plant," Revision 3

40. Harris Nuclear Plant, Plant Operating Manual Volume 3, Part 4, Emergency Operating Procedure EOP-ECA-0.0, "Loss of All AC Power," Revision 3

41. FSG-011, "Alternate SFP Makeup and Cooling," Revision 0

42. FSG-005, "Initial Assessment and FLEX Equipment Staging," Revision 0

43. Engineering Change EC 91701, "Seismically Upgrade SFP Piping for NTTF 4.2 (FLEX)," Revision 0

44. EC 91691, "Analysis of Delivery Path of FLEX Equipment (Item 30)," Revision 1

45. CSD-EG-HNP-8888, "Flexible Response to Extended Loss of All AC Power (FLEX)," Revision 0

46. Letter from Jack Davis (NRC) to Jack Stringfellow (PWROG) "Boron Mixing Endorsement Letter in Regards to Mitigation Strategies Order EA-12-049", dated January 8, 2014 (ADAMS Accession No. ML13276A183)

47. HNP-M/FLEX-0006, "FLEX AFW Pump Sizing and Pressure Drop Calculation," Revision 1

48. EC91680, "FLEX RCS Injection Evaluation," Revision 5

49. EC 91683, "Diesel Fuel Oil Transfer Evaluation for NTT 4.2 (FLEX)," Revision 0

50. EC 91684, "Calculation of Available Volume from Condensate Storage Tank (Seismic and Normal)," Revision 0

51. FSG-006, "Alternate CST Makeup," Revision 0

52. FSG-003, "FLEX Low Pressure Feedwater," Revision 0

53. Letter from B. Waldrep (Duke), "Carolina Power and Light Company's Overall Integrated Plans in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order No. EA-12-051)," dated February 28, 2013 (ADAMS Accession No. ML13086A096)

54. Letter from Araceli T. Biloch Colon (NRC) to Ernie Kapopoulos (Progress Energy) regarding Shearon Harris Nuclear Power Plant, Unit 1 – Request for Additional Information Regarding Overall Integrated Plan for Reliable Spent Fuel Pool instrumentation (Order EA-12-051), dated July 11, 2013 (TAC No. MF0792) (ADAMS Accession No. ML13189A225)

55. Letter from E. Kapopoulos (Duke), "Response to Request for Additional Information Regarding Overall Integrated Plan for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 12, 2013 (ADAMS Accession No. ML13225A494)

56. Letter from Andrew Hon (NRC) to Ernie Kapopoulos (Duke) regarding Shearon Harris Nuclear Power Plant, Unit 1 – Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan for Implementation of Order EA-12-051, Reliable Spent Fuel Pool Instrumentation (TAC No. MF0792), dated November 19, 2013 (ADAMS Accession No. ML13280A482)

57. Calculation HNP-F/NFSA-0240, Revision 1, "HNP Boration During Extended Loss of Power (ELAP) Event," dated March 5, 2015

58. Letter from Andrew Hon (NRC) to Ernie Kapopoulos (Duke), "First Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 26, 2013 (ADAMS Accession No. ML13242A010)

59. Letter from Sean T. O'Connor (Duke), "Second Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses with regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 27, 2014 (ADAMS Accession No ML14063A198)

60. Letter from John D. Dufner (Duke), "Third Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated August 22, 2014 (ADAMS Accession No. ML14251A015)

61. Letter from Benjamin C. Waldrep (Duke), "Fourth Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," dated February 23, 2015 (ADAMS Accession No. ML15055A107)

62. Letter from John Boska (NRC) to Steven D. Capps (Duke Energy) regarding McGuire Nuclear Station, Units 1 and 2 – Report for the Onsite Audit of AREVA Regarding Implementation of Reliable Spent Fuel Pool Instrumentation Related to order EA-12-051 (TAC Nos. MF1062 AND MF1063), dated September 15, 2014 (ADAMS Accession No. ML14203A326)

63. Calculation Number HNP-M-FLEX-0008, "ESW Pump and Hose Sizing for FLEX Strategies," Revision 0

64. NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195)

65. Pressurized-Water Reactor Owners Group (PWROG) PWROG-14064-P, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances," Revision 0

66. WCAP-10541-P, "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Revision 2

67. Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-14-1, dated February 10, 2014

68. Calculation Number E4-0008, "125VDC 1E Battery Sizing and Battery/Panel Voltages for Station Blackout," Revision 7

69. FSG-004, "ELAP DC Bus Load Shed/Management", Revision 0

70. Letter from Nicholas Pappas, Nuclear Energy Institute to Jack R. Davis, NRC regarding "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," dated August 27, 2013 (ADAMS Accession No. ML13241A186)

71. Letter from Jack R. Davis, NRC to Joseph Pollock Nuclear Energy Institute, regarding NRC staff endorsement of "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," dated September 16, 2013 (ADAMS Accession No. ML13241A188)

72. Calculation HNP-M/FLEX-0012, "Harris Reactor Auxiliary Building Extended Loss of ac Power FLEX Response", Revision 2
73. EC 91688, "ESW to SSE FP Hose Rack Pressure and Flow Evaluation," Revision 0

74. Engineering Change 91686, "Containment Temperature and Pressure Evaluation for NTTF 4.2 FLEX," Revision 1

75. HNP-M/FLEX-0009, "FLEX ELAP Containment Temperature Pressure Analysis", Revision 0

76. FSG-012, "FLEX Containment Cooling," Revision 0

77. EC 91690, "Loss of HVAC Analysis," Revision 0

78. HPN/FLEX-0013, "Harris Fuel Handling Building Extended Loss of AC Power FLEX Response," Revision 0

79. PWROG-14074-P, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power," Revision 0

80. Engineering Change (EC) 89579, "Fukushima Response Project – SFP Wide Range Level Indication – HNP," Revision 0

81. Sloshing Analysis HNP-C/FLEX-0003, "Seismic Induced Hydraulic Response in the Harris Spent Fuel Pool," Revision 0
82. Calculation 32-9221971-000, "Qualification of Difference in as-tested and as-fabricated Mounting Configuration for VEGA Power Control Panel Mounting Assembly," Revision 0
83. Drawing SK-89579-C-1001, Sheet 1 thru 4, "SFPLI Wave Guide Pipe Supports," Revision 0
84. Drawing SK-89579-C-1002, Sheet 1 thru 6, "Power Control Panel Assembly Mounting Locations, Sections, and Details," Revision A
85. Drawing SK-89579-C-1003 (Areva Drawing 02-9214553B), "Horn and Sensor Support SFP A," Revision 0
86. Drawing SK-89579-C-1004 (Areva Drawing 02-9214554B), "Horn and Sensor Support SFP B," Revision 0
87. Drawing SK-89579-C-1005(Areva Drawing 02-9214555B), "Horn and Sensor Support SFP C," Revision 0

88. Drawing Markup CAR-2165 G-022, "General Arrangement Fuel Handling Building Plants – Sheet 1," Revision 0

89. Calculation NAI-1725-003, "GOTHIC Verification and Sensitivity Studies for Predicting Hydrodynamic Response to Acceleration in Rectangular Shaped Pools," Revision 0

90. Westinghouse LTR-FSE-13-46, Revision 0, P-Attachment, "Westinghouse Response to NRC Generic Request for Additional Information (RAI) on Boron Mixing in Support of the Pressurized Water Reactor Owners Group (PWROG)," dated August 16, 2013 (Adams No. ML13235A135) (Enclosure withheld from Public disclosure for Proprietary Reasons)

91. Calculation Number E-6009, "Fukushima Diesel Generator Sizing Calculation," Revision 1

92. Letter from Frankie Vega (NRC) to Benjamin C. Waldrep, Shearon Harris Power Plant, Unit 1, Staff Assessment of Information Provided Pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident (TAC No. MF3952), dated December 18, 2015 (ADAMS Accession No. ML15349A149)

93. Procedure AP-300, "Severe Weather Response," Revision 18

94. Letter, Duke Energy, "Supplement to Notification of Full Compliance with Order EA-12-049, 'Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events' and with Order EA-12-051, 'Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation' Shearon Harris Nuclear Power Plant, Unit 1," dated June 29, 2016 (ADAMS Accession No. ML16182A047)

95. HPN/FLEX-0016, "Harris Fuel Handling Building Alternative Cooling," Revision 0

96. WCAP-17601-P, "Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs," Revision 1

97. Letter from Stephen Wyman (NRC) to all Power Reactor Licensees regarding "Staff Review of High Frequency Confirmation Associated with Reevaluated Seismic Hazard in Response to March 12, 2012 50.54(f) Request for Information," dated February 18, 2016 (ADAMS Accession No. ML15364A544)

Principal Contributors: Austin Roberts
Kerby Scales
Khoi Nguyen
Bruce Heida
Kevin Roche
Stephen Monarque
Barry Smith

Date: August 30, 2016

If you have any questions, please contact Stephen Monarque, Orders Management Branch, HNP Project Manager, at 301-415-1544 or at Stephen.Monarque@nrc.gov.

Sincerely,

/RA/

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

Docket No.: 50-400

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

DISTRIBUTION:

PUBLIC
JLD R/F
RidsNrrDorlLpl2-2 Resource
RidsNrrLASLent Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn2MailCenter Resource
SMonarque, NRR/JLD
RidsNrrPMShearonHarris Resource
KRoche, NRR/JLD
KScales, NRR/JLD
JMiller, NRR/JLD
KNguyen, NRR/JLD

ADAMS Accession No. ML16217A449 ***via email**

OFFICE	NRR/JLD/JOMB/PM	NRR/JLD/LA	NRR/JLD/JERB/BC	NRR/JLD/JCBB/BC
NAME	SMonarque	SLent	SBailey*	JQuichocho*
DATE	08/09/2016	08/09/2016	08/16/2016	08/17/2016

OFFICE	NRR/JLD/JOMB/BC(A)		
NAME	MHalter		
DATE	8/30/16		

OFFICIAL AGENCY RECORD