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

October 20, 2016

Mr. Robert T. Simril
Site Vice President
Duke Energy Carolinas, LLC
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 (CAC NOS. MF1162, MF1163, MF1060, AND MF1061)

Dear Mr. Simril:

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

By letter dated February 28, 2013 (ADAMS Accession No. ML13066A173), Duke Energy Carolinas, LLC (Duke, the licensee) submitted its OIP for Catawba Nuclear Station (Catawba or CNS), Units 1 and 2, in response to Order EA-12-049. At six month intervals following the submittal of its OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation (SE). By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated October 28, 2013 (ADAMS Accession No. ML13281A562), and February 20, 2015 (ADAMS Accession No. ML15035A679), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated May 1, 2015 (ADAMS Accession Nos. ML15126A277), Duke submitted a compliance letter for CNS, Unit 2 in response to Order EA-12-049. By letter dated February 15, 2016 (ADAMS Accession No. ML16049A041), Duke submitted a compliance letter for CNS, Unit 1 and Final Integrated Plan (FIP) for CNS, Units 1 and 2, in response to Order EA-12-049. The compliance letters stated that the licensee had achieved full compliance with Order EA-12-049 for each unit, as applicable. By letters dated March 31, 2016 (ADAMS Accession No. ML16095A208) and September 27, 2016 (ADAMS Accession No. ML16273A303), the licensee submitted supplemental information regarding compliance with Order EA-12-049 at CNS.

By letter dated February 28, 2013 (ADAMS Accession No. ML13086A095), Duke submitted its OIP for CNS in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached SE. By letters dated October 28, 2013 (ADAMS Accession No. ML13281A562), and February 20, 2015 (ADAMS Accession No. ML15035A679), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated May 1, 2015 (ADAMS Accession Nos. ML15126A277), Duke submitted a compliance letter for CNS, Unit 2, in response to Order EA-12-051. By letter dated February 15, 2016 (ADAMS Accession No ML16049A041), Duke submitted a compliance letter for CNS, Unit 1, in response to Order EA-12-051. The compliance letters stated that the licensee had achieved full compliance with Order EA-12-051 for each unit, as applicable. By letter dated March 31, 2016 (ADAMS Accession No. ML16095A208), the licensee submitted supplemental information regarding compliance with Order EA-12-051 at CNS.

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

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

Sincerely,

Mandy K Halter

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

Docket Nos.: 50-413 and 50-414

Enclosure:
Safety Evaluation

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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 CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

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

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

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

2.0 REGULATORY EVALUATION

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

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

2.1 Order EA-12-049

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

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

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

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

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

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

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

2.2 Order EA-12-051

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 [Reference 10], Duke Energy Carolinas, LLC (Duke, the licensee) submitted its Overall Integrated Plan (OIP) for Catawba Nuclear Station (Catawba, CNS) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 28, 2014 [Reference 12], August 28, 2014 [Reference 13], February 26, 2015 [Reference 14], and August 26, 2015 [Reference 15], the licensee submitted six-month updates to its OIP. By letter dated August 28, 2013 [Reference 16], the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 33]. By letters dated February 6, 2014 [Reference 17] and February 20, 2015 [Reference 18], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated May 1, 2015 [Reference 39], the licensee reported that CNS, Unit 2 had reached compliance with the requirements of Order EA-12-049. By letter dated February 15, 2016 [Reference 19], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved for CNS, Unit 1, and submitted a Final Integrated Plan (FIP). By letters dated March 31, 2016, and September 27, 2016 [References 40 and 58], the licensee provided supplemental information regarding compliance with Order EA-12-049 at CNS.

3.1 Overall Mitigation Strategy

Attachment 2 to Order EA-12-049 describes the three-phase approach required for mitigating BDBEEs in order to maintain or restore core cooling, containment and SFP cooling capabilities. The phases consist of an initial phase (Phase 1) using installed equipment and resources, followed by a transition phase (Phase 2) in which portable onsite equipment is placed in service, and a final phase (Phase 3) in which offsite resources may be placed in service. The timing of when to transition to the next phase is determined by plant-specific analyses.
While the initiating event is undefined, it is assumed to result in an extended loss of ac power (ELAP) with a loss of normal access to the UHS. Thus, the ELAP with a loss of normal access to the UHS is used as a surrogate for a BDBEE. The initial conditions and assumptions for the analyses are stated in NEI 12-06, Section 3.2.1, and include the following:

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

The CNS site hosts two Westinghouse pressurized-water reactors (PWRs) with ice condenser containments. The FIP describes the licensee's three-phase approach to mitigate a postulated ELAP event affecting these reactors.

Initially, both reactors are assumed to be at full power. In response to the ELAP event, the reactor and turbine at each unit would trip. Following these trips, the turbine-driven auxiliary feedwater (TDAFW) pump would supply water to the secondary side of the steam generators (SGs) to absorb residual heat from the reactor coolant system (RCS). This heat transfer would generate secondary steam, which would be vented to the atmosphere via main steam safety valves or SG power-operated relief valves (PORVs). According to the licensee's FIP, plant operators would conduct an RCS cooldown using the SG PORVs starting within 0.6 hours of ELAP event initiation. The SGs would be depressurized to approximately 240 pounds per square inch gauge (psig), with the RCS cooldown rate being limited to less than 100 degrees Fahrenheit (°F)/hour. Holding SG pressure at 240 psig is intended to prevent nitrogen injection from the cold leg accumulators (CLAs) into the RCS.

The TDAFW pump suction will be supplied from the upper surge tanks (USTs) in the condensate storage system, the auxiliary feedwater condensate storage tank (CACST), or the condenser hotwell, which are condensate grade water sources. However, these sources are susceptible to damage from wind and seismic events and may not be available. The TDAFW pump suction can also be supplied from buried piping in the condenser circulating water system, which has been evaluated to be seismically robust and can supply sufficient water to support decay heat removal for at least 48 hours. The valve to enable the TDAFW pump to take suction from embedded circulating water piping automatically opens at a pre-determined setpoint such that a suction source is maintained. Operators will control SG level by throttling auxiliary feedwater system flow control valves from the control room. If necessary, throttling of these control valves (or other system valves) can also be accomplished locally.

The vital station batteries provide dc power for essential instrumentation. Vital battery load shedding will be initiated 2.5 hours into the event. The CNS load shedding strategy will maintain power supply from the vital batteries until the associated battery chargers can be re-powered.

The Phase 2 core cooling strategy continues to use the SGs as the heat sink. Catawba has multiple strategies for providing feedwater to the SGs using FLEX equipment, including drawing water from the standby nuclear service water pond (SNSWP – the UHS) and the FLEX raw water distribution system. Personnel will deploy 600 Volt alternating current (Vac) diesel generators (DGs) to provide power for the FLEX strategies within 9 hours of the event. The 600 Vac FLEX DGs will be connected to the plant motor control centers (MCCs) for re-powering via the FLEX "Backbone," which consists of permanently installed cables, portable panelboards, and transformers.

Operators will provide Phase 2 RCS makeup using a portable high-pressure pump with suction provided from the refueling water storage tank (FWST). Sufficient borated water will be added to maintain the core sub-critical, in a xenon-free condition, at 350°F.

The CLA isolation valves, re-powered using the FLEX DGs, will be closed to prevent nitrogen from being injected into the RCS prior to depressurizing the SGs to 160 psig.

A portable FLEX sump pump is placed in each TDAFW pump pit to pump out the room sump before flooding impacts operation of the TDAFW pump, which could occur as soon as 7 hours into the event. Additionally, the groundwater drainage system sump pumps must be in operation by 10.6 hours into the event to prevent installed sump pump motors from being flooded. Operators can also deploy FLEX sump pumps in various locations (in addition to the TDAFW pump pits) to manage groundwater intrusion.

In Phase 3, after verifying adequate RCS boron concentration to support an additional cooldown, plant operators would further depressurize the SGs into the range of 80-100 psig to support long-term integrity of the reactor coolant pump (RCP) seals. Operators may further transition to core cooling using the residual heat removal (RHR) system. Should this course be pursued, operators would repower the component cooling water pumps to provide cooling water to the RHR system. The National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) will deliver equipment to CNS to allow for repowering the RHR and the component cooling water systems. In addition to cooling the reactor core using heat exchangers, the RHR pumps can also provide borated makeup water to the core by taking suction from the FWST. If needed, the licensee's FIP states that mobile boration units from the NSRC can be used to refill the FWST through existing vents. By letter dated September 27, 2016 [Reference 58], the licensee stated that, alternatively, discharge flow from the mobile boration units could be used to refill a portable suction tank should the FWST be damaged in such a way as to prevent refilling. The prepared coolant in the portable suction tanks could then be injected into the RCS using FLEX pumps. The NSRC will deliver a water treatment skid that can provide a cleaner water source than the SNSWP. Additional diesel fuel for portable equipment will be brought in from off-site resources when required.

Regarding SFP cooling, in its FIP the licensee stated that no actions are required during Phase 1 for SFP make-up because the time to boil is sufficient to enable deployment of Phase 2 equipment. For the worst case heat load scenario, the SFP will begin to boil after an ELAP/loss of normal access to the UHS event in 8.8 hours. This scenario assumes a maximum initial SFP temperature of 125°F and a full core offload during an outage after 6 days. During normal operation (initial SFP temperature of 125°F and 21 days after the beginning of a refueling outage), SFP boiling will begin in 37.0 hours and level can be maintained at least 10 feet above the top of the fuel for 202.9 hours with no makeup. In Phase 1, operators will monitor SFP water level using SFP level instrumentation.

To compensate for SFP boil-off later in the event in Phase 2, operators will provide makeup water by pumping raw water from the SNSWP using the FLEX low pressure pump. The flow path will use nuclear service water system and SFP cooling system piping either directly to the SFP or alternately with a hose connection from nuclear service water piping to the SFP cooling system skimmer loop.

Long-term SFP cooling in Phase 3 will be accomplished by re-powering the installed SFP cooling and component cooling water pumps using a portable generator obtained from the NSRC to provide cooling via normal means. The heat sink for the component cooling water system will be from the nuclear service water system that is being supplied from the portable diesel-driven pump located at the SNSWP.

Regarding containment cooling and integrity, in its FIP the licensee stated that ice in the ice condenser initially serves as a heat sink for the containment atmosphere. Once the lower containment pressure rises sufficiently to open the ice condenser doors, steam escaping the primary and/or secondary systems would be condensed as it rises from lower containment through the ice condenser and into upper containment. A containment analysis demonstrates that containment pressure is expected to remain below the design pressure of 15 psig during Phase 1 without operator actions.

Operators will use the FLEX electrical distribution system to enable Phase 2 actions to maintain containment integrity. These actions include starting a hydrogen skimmer fan within 24 hours of event occurrence to limit the temperature increase in the SG and pressurizer compartments and repowering hydrogen igniters to prevent a buildup of hydrogen in case the ELAP event degrades to core damage

Following deployment of an NSRC 480 Vac generator, a 480 to 600 Vac step-up transformer, and energizing of the 600 Vac MCCs in Phase 3, operators will start two lower containment ventilation units (LCVUs) within 48 hours to limit the temperature increase in the SG and pressurizer compartments. Additionally, one containment air return fan (CARF) will be started within 52 hours of the event to establish an air flow path through the ice condenser, reduce containment pressure, and limit further heat-up of the SG and pressurizer compartments. Operators will complete transition to RHR system cooling and cool down to Mode 5 (i.e., average RCS temperature ≤ 200°F) within 6 days of ELAP initiation to prevent challenging containment temperature and pressure limits following ice bed depletion.

Below are specific details on the licensee's strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of an ELAP, and the results of the staff's review of these strategies. The NRC staff evaluated the licensee's strategies against the endorsed guidance in NEI 12-06, Revision 0.

3.2 Reactor Core Cooling Strategies

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

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 to the heat sink via natural circulation. 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/loss of normal access to the UHS event presumes that, per endorsed guidance from NEI 12-06, both units would have been operating at full power prior to the event. Therefore, the SGs may be credited as the heat sink for core cooling during the ELAP/loss of normal access to the UHS 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/loss of normal access to the UHS event are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions where one or more units are shut down or being refueled is reviewed separately in Section 3.11 of this evaluation.

3.2.1 Core Cooling Strategy and RCS Makeup

3.2.1.1 Core Cooling Strategy

3.2.1.1.1 Phase 1

As stated in the licensee's FIP dated February 15, 2016 [Reference 19], the heat sink for core cooling in Phase 1 would be provided by the four SGs at each unit, which would be fed simultaneously by the unit's TDAFW pump. The TDAFW pump would preferentially take suction from several sources of condensate-grade feedwater:

- the two USTs in the condensate storage system, which have a combined capacity of 85,000 gallons per unit
- the CACST, which has a capacity of 42,500 gallons
- the main condenser hotwell, which has a capacity of 170,000 gallons

However, none of these sources is fully robust to all applicable external hazards. As such, the credited Phase 1 source of feedwater for an ELAP/loss of normal access to the UHS scenario is the buried piping in the condenser circulating water system. The underground circulating water piping will be automatically aligned to the TDAFW pump suction if the suction pressure drops below a pre-determined setpoint.

The licensee's Phase 1 strategy directs the initiation of a cooldown and depressurization of the RCS within two hours of the initiation of the ELAP/loss of normal access to the UHS event. Over a period of approximately 2 hours, the licensee would gradually cool down the RCS from post-trip conditions until a SG pressure of 240 psig is reached, which results in an RCS cold-leg temperature of approximately 403°F. Cooldown and depressurization of the RCS significantly extends the expected coping time under ELAP/loss of normal access to the UHS conditions because it: (1) reduces the potential for damage to RCP seals (as discussed in Section 3.2.3.3), and (2) allows borated coolant stored in the nitrogen-pressurized cold leg accumulators to passively inject into the RCS to offset system leakage and add negative reactivity. To remove heat, operators will discharge steam through the SG PORVs, which can be operated from the control room using vital battery power and safety-grade nitrogen backup for the valve actuators. The SG PORVs can also be manually operated. The licensee has calculated that each PORV is provided with sufficient nitrogen to supply two full strokes of the valve and 8 hours of expected system leakage. The licensee expects that the ELAP event should require less than two full strokes of the valve in the first 6 hours. However, as discussed during the audit, operators would eventually need to manually actuate the SG PORVs using hand wheels, hook up more nitrogen bottles, or establish another air supply.

By terminating the initial RCS cooldown at a SG pressure of 240 psig, the licensee has determined that injection of the nitrogen cover gas in the accumulators into the RCS will be prevented. The NRC staff's audit found that the methods used in the licensee calculations for this determination to be appropriate.

3.2.1.1.2 Phase 2

The licensee's FIP states that the primary strategy for core cooling in Phase 2 would be to continue using the SGs as a heat sink. During Phase 2, operators can deploy portable FLEX pumps to either provide long-term supply to the TDAFW pump (if SG pressure is high enough to support its continued use) or inject directly to the SGs (if the TDAFW pump can no longer be used):

- If the TDAFW pump is available, a portable, low-pressure, diesel-driven FLEX pump will take suction from the SNSWP and discharge to the nuclear service water system. Operators will align flow from the nuclear service water system to the suction of the TDAFW pump.

- If the TDAFW pump is not available due to low steam pressure, a portable, low-pressure, diesel-driven FLEX pump will take suction from the SNSWP and pressurize a raw water distribution header. A portable, medium-pressure, diesel-driven FLEX pump will be deployed to take suction from the distribution header and discharge to a connection in the interior and exterior SG Doghouse, feeding the SGs at a rate of up to 300 gpm, given an SG pressure of 300 psig. Operators can establish the necessary flow path using only hoses, or using a combination of hoses and the fire protection system piping. This method establishes an essentially unlimited source of secondary makeup.

- A less-preferable option, available as a contingency, would be to inject from the SNSWP with the low-pressure FLEX pump directly to the SGs, without the medium-pressure FLEX pump. This method has less capacity than using the medium-pressure pump and entails feeding and steaming SGs that are nearly dry. The 150-psig discharge pressure of this pump would require lowering SG pressure well below the 240-psig cooldown terminus established for the analyzed ELAP event, which increases the risk of injecting nitrogen from the CLAs into the RCS.

The transition from the TDAFW pump to a FLEX pump will be executed before reactor core decay heat diminishes to the point that SG pressure cannot be maintained at the minimum value necessary to support TDAFW pump operation. For both the TDAFW pump supply and FLEX SG feed methods, primary and alternate connections exist which are appropriately protected from applicable hazards.

In addition to the primary core cooling strategy discussed above, the licensee indicated that a number of other water sources may be available. In particular, the USTs, CACSTs, and condenser hotwell have condensate-grade water which will not foul the SGs; however, these sources are not protected from all external hazards and have not been credited. The licensee's FIP states that raw water from the SNSWP may be used as SG feed water for a limited duration, but that water purification equipment from the NSRC would be preferentially placed into service when it becomes available in Phase 3.

The FIP also states that portable FLEX DGs, power distribution panels, and cables will be placed into service in Phase 2, within 9 hours of the initiating event, to provide electrical power to equipment supporting the core cooling strategy, including the accumulator isolation valves. Once RCS boration is complete (see Section 3.2.1.2 below), operators will shut the accumulator isolation valves to prevent injection of their nitrogen cover gas when the RCS is further cooled and depressurized.

3.2.1.1.3 Phase 3

According to its FIP, the licensee's initial Phase 3 core cooling strategy continues to use the SGs as the heat sink, with additional offsite equipment and resources placed into service. CNS will receive water purification equipment and a mobile boration skid from the NSRCs to ensure a long-term source of purified water. Catawba will also obtain additional diesel fuel from off-site sources for continued operation of diesel-powered equipment, if necessary. The NSRCs will provide two 1-megawatt (MW) DGs per unit, which will allow repowering a 4 kilovolt essential bus and required load centers. This increase in FLEX electrical power capacity will enable the repowering of specific installed plant equipment.

In Phase 3, the licensee would further cool down and depressurize the RCS. As described by letter dated September 27, 2016 [Reference 58], the licensee intends to depressurize the SGs to 160 psig at approximately 25 hours into the analyzed ELAP event. The licensee stated that this SG depressurization would result in an RCS cold leg temperature of approximately 371°F. After conducting additional preparations to support a further temperature and pressure reduction, including establishing flow from the FLEX SG makeup pumps and providing low-temperature overpressure protection for the RCS, the licensee would implement an additional SG depressurization into the range of 80-100 psig. The licensee stated that implementation of this additional depressurization, including supporting activities, could be completed within approximately 37 hours from the initiation of the ELAP event.

In its FIP, the licensee further stated that CNS can transition to Phase 3 core cooling using the RHR system after RCS cold leg temperature has been reduced below 350°F and RCS pressure has been reduced below 385 psig. Transitioning to RHR would be expected to reduce RCS leakage into containment, thereby managing excessive containment conditions (temperature, pressure, and sump level). The NSRC will provide two low-pressure pumps (over 3000 gallons per minute (gpm) each), which will take suction from the SNSWP and be connected to the nuclear service water system to deliver cooling water to the component cooling water heat exchangers (one for each unit). Portable diesel generators, also furnished by the NSRC, will power the installed component cooling water and RHR pumps.

3.2.1.2 RCS Makeup Strategy

3.2.1.2.1 Phase 1

Under ELAP conditions, RCS inventory will tend to diminish gradually due to leakage through RCP seals and other leakage points. Furthermore, the initial RCS cooldown starting at two hours into the event would result in a significant contraction of the RCS inventory, to the extent that the pressurizer would drain and a vapor void would form in the upper head of the reactor vessel. As is typical of operating PWRs, prior to implementing the Phase 2 FLEX strategy, CNS does not credit active RCS makeup. However, passive injection from the nitrogen-pressurized accumulators would occur as the RCS is depressurized below the accumulator cover gas pressure, which would result in the addition of borated coolant to the RCS. As discussed further below, the licensee has determined that: (1) sufficient reactor coolant inventory would be available throughout Phase 1 to support heat transfer to the SGs via natural circulation without crediting the active injection of RCS makeup, and (2) according to the core operating history specified in NEI 12-06, a sufficient concentration of xenon-135 should exist in the reactor core to ensure sub-criticality throughout Phase 1, considering the planned cooldown profile.

3.2.1.2.2 Phase 2

In order to maintain sufficient borated RCS inventory in Phase 2, the licensee states that a diesel-driven high-pressure FLEX pump would be deployed to inject borated coolant from the FWST into the RCS. The licensee will commence RCS makeup and boration no later than 11.6 hours after the start of the ELAP event, in order to ensure reactivity control and prevent the onset of reflux cooling. The primary path for borated makeup to the RCS is through FLEX connections on the safety injection system; fully protected connections are available on both the "A" and "B" safety injection trains, representing primary and alternate paths.

The FWST is seismically qualified but is potentially vulnerable to a wind-borne missile strike above its 14-foot-high protective wall. Given the worst-case missile strike above the wall, the FWST would still retain approximately 105,000 gallons. This represents sufficient inventory for boration and RCS injection throughout Phase 2 and well into Phase 3.

3.2.1.2.3 Phase 3

In Phase 3, the RCS makeup strategy is a continuation of the Phase 2 strategy, supplemented as needed with equipment provided by the NSRC. The FIP states that, beyond 72 hours, operators would be able to replenish the volume of the FWST as necessary using an NSRC-supplied mobile boration skid, or with borated water transported by truck from McGuire Nuclear Station. If the FWST could not be refilled due to damage (e.g., from a tornado-driven missile strike), then the licensee could align the discharge of the mobile boration skid to portable mixing and suction tanks. The prepared borated coolant in the portable suction tanks could then be injected into the RCS using FLEX pumps. Additional powdered boric acid would also be supplied by the NSRC. The NRC staff notes that the licensee should begin using purification equipment from the NSRC as soon as practical considering the overall event response prioritization and the necessity to facilitate the use of higher quality water for RCS makeup.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

In the FIP, Section 2.8.1, the licensee states that an external flooding event from extreme precipitation or dam failure may inundate portions of the site, but that flood waters will recede from all planned FLEX deployment paths and staging areas so that all time constraints will still be met. Therefore, there are no variations to the core cooling strategy in the event of a flood.

3.2.3 Staff Evaluations

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

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

3.2.3.1.1 Plant SSCs

Core Cooling

The licensee provided descriptions in its FIP for the permanent plant SSCs to be used to support core cooling for Phase 1 and 2. The Seismic Category I structures include Containment, Auxiliary Building, Nuclear Service Water System Pumphouse, SG Doghouses, and the Fuel Handling Building, which are all protected from all external hazards. The licensee identified in the FIP the safety-related, or seismically robust, piping from the following systems are credited as part of the FLEX strategy and protected from all external events: RCS, auxiliary feedwater system, safety injection system, RHR system, the nuclear service water system, the component cooling water system, the condenser circulating water system, the main steam to auxiliary equipment system, the main steam vent to atmosphere system, the SG wet layup recirculation system, the SFP cooling system, and the refueling water system. The licensee also described the TDAFW pumps, which are used to supply feedwater to the SGs during Phase 1, as being located in the Auxiliary Building and protected from all external hazards. Additionally, the licensee described in the FIP six groundwater drainage system sump pumps, which will be re-powered by the 600V FLEX DGs. The sump pumps will be necessary to address potential flooding near the TDAFW pump pits about 10.6 hours into the ELAP event. The "A "and "B" sump pumps (4 pumps) are protected from all external hazards. The "C" sump pumps (2 pumps) will be used if available despite not being fully protected. The SG PORVs are used to remove heat during SG cooling, and they are located inside the SG Doghouses, which are Seismic Category I structures.

The licensee described two water sources that are credited as part of the FLEX strategy after ELAP in the FIP. The condenser circulating water system is described as the primary preferred source of SG feedwater because of its protection from all external hazards. The Unit 1 condenser circulating water system has approximately 982,000 gallons of total usable water and the Unit 2 condenser circulating water system has approximately 1,280,000 gallons of total usable water. The licensee estimated that with both units, water from the condenser circulating water system can support the core cooling functions for at least 48 hours. The long-term SG makeup water source is described in the FIP as the SNSWP, which holds water capable for indefinite coping at CNS. The SNSWP is fully protected from all external hazards and can also be used when the water from the condenser circulating water system is depleted for Phase 3. The licensee also discussed in its FIP that additional water sources such as the USTs and the hotwell would be used if available, however they are not protected from external hazards.

Based on the design and locations of the TDAFW pumps and PORVs, the redundancy of the sump pumps, the condenser circulating water system in both units, and the SNSWP as described in the FIP, the NRC staff finds that the plant SSCs and water sources should be available to support core cooling during an ELAP caused by a BDBEE, consistent with Condition 4 of NEI 12-06, Section 3.2.1.3. Additionally, due to the design and locations of the primary and alternate auxiliary feedwater connection points, as described below in Section 3.7.3.1 and in the FIP, at least one of the connection points should be available to support core cooling through a portable FLEX pump during an ELAP caused by a BDBEE, consistent with NEI 12-06, Section 3.2.2 and Table D-1.

RCS Inventory Control

The licensee described in its FIP that the FWSTs are the credited source of borated water for reactivity control and RCS makeup. The credited minimum inventory of the FWSTs is 377,537 gallons. The FWST is seismically-qualified and the bottom portion is protected by a missile wall. The licensee stated that RCS makeup would be provided by the FWST for the duration of the ELAP event as long as the top portion remains intact. The licensee also stated that makeup to the FWST would be required within 52 hours if the top portion is damaged and will be made up from the NSRC mobile boration skid or borated water trucked in from off-site sources, such as McGuire. The boric acid tank (BAT) and the borated water in the FWST annulus are also available for RCS makeup as needed.

Based on the design and availability of the FWST and BAT as described in the FIP, the NRC staff finds that a borated water source should be available to support RCS inventory control during an ELAP caused by a BDBEE, consistent with Condition 3 of NEI 12-06, Section 3.2.1.3. Additionally, due to the design and location of the primary and alternate RCS injection connection points, as described below in Section 3.7.3.1 and in the FIP, at least one of the connection points should be available to support RCS injection through the FLEX pump during an ELAP caused by a BDBEE, consistent with NEI 12-06, Section 3.2.2 and Table D-1.3.2.3.1.2.

3.2.3.1.2 Plant Instrumentation

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

- TDAFW pump flow
- SG water level (narrow range)
- SG pressure
- RCS hot-leg temperature
- RCS pressure (wide range)
- core exit thermocouples
- pressurizer level
- reactor vessel level indicating system
- neutron flux
- FWST level
- RCP seal leakoff flow

These instruments are initially powered by vital station batteries. In Phase 2, long-term power is established for these essential instruments by recharging station batteries from a portable FLEX DG. Alternatively, operators can directly re-power instrument cabinets using smaller portable generators. Based on this information provided by the licensee, the NRC staff understands that indication for the above instruments would be available and accessible continuously throughout the ELAP event.

The primary monitoring strategy for all of these parameters is to obtain readings from the main control room (MCR). If this is not possible, operators can monitor the feedwater flow rate and SG pressure locally; other instruments can be read from inside the Process Control System 7300 cabinet in the control room using portable test equipment per a FLEX Support Guideline (FSG), FSG-07, "Loss of Vital Instrumentation or Control Power."

The instrumentation available to support the licensee's strategies for core cooling and RCS inventory during the ELAP event is consistent with and in some cases exceeds the recommendations specified in the endorsed guidance of NEI 12-06. In particular, the staff noted that the availability of RCP seal leakoff flow instrumentation during an ELAP/loss of normal access to the UHS exceeds the recommendations of NEI 12-06. Although availability of this indication may be beneficial in providing insight into the RCP seal leakage behavior, the NRC staff observed that typical leakoff line flow instruments are designed for single-phase liquid; whereas, during the ELAP event, the two-phase flow expected through the RCP seals may result in a quantitative indication that is not representative of the actual flowrate.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee 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 NRC's 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 (such as CNS), the reflux cooling mode is said to exist when vapor boiled off from the reactor core flows out the saturated, stratified RCS hot legs and condenses in the SG tubes, with the majority of the condensate subsequently draining back into the reactor vessel through the hot legs in countercurrent fashion. Quantitatively, as reflected in documents such as the PWR Owners Group (PWROG) report PWROG-14064-P, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances," Revision 0, 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 [Nuclear Steam Supply System] Designs." Subsequently, a series of additional simulations performed by the 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 safety evaluation (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 staff was unable to conclude that the generic analysis performed in WCAP-17601-P could be directly applied to all Westinghouse PWRs, as the vendor originally intended. In PWROG-14064-P, Revision 0, the industry subsequently recognized that the generic analysis would need to be scaled to account for plant-specific variation in RCP seal leakage. However, the staff's review, supported by sensitivity analysis performed with the TRACE code, further identified that plant-to-plant variation in additional parameters, such as RCS cooldown terminus, accumulator pressure and liquid fraction, and initial RCS mass, could also result in substantial differences between the generically predicted reference coping time and the actual coping time that would exist for specific plants.

As identified in the licensee's FIP, CNS relies upon the generic thermal-hydraulic analysis documented in Sections 5.2.1 and 5.2.2 of WCAP-17601-P. During the audit, the staff reviewed key plant-specific parameters for CNS relative to those assumed in the analysis from WCAP-17601-P to ensure applicability of the analysis. A comparison of parameters associated with the safety injection accumulators indicated that the passive accumulator injection expected for CNS would likely be less than what was observed in the generic analysis. However, the NRC staff judged that this discrepancy would be overcome by a favorable difference of larger magnitude in initial RCS volume for CNS, Unit 1. Consequently, the staff's audit considered the generic plant design parameters assumed in Sections 5.2.1 and 5.2.2 of WCAP-17601-P as being applicable to CNS, Unit 1 from an overall perspective. However, this is not the case for CNS, Unit 2, as a reduced RCS volume, along with reduced passive accumulator injection, would be expected to result in less coping time as compared to the generic analysis. Thus, CNS, Unit 2 is not adequately bounded by the generic parameters assumed in WCAP-17601-P.

The analyses from WCAP-17601-P were subsequently analyzed further by the PWROG, and estimated times to enter reflux cooling were documented in 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". The licensee's strategy to provide RCS makeup by 11.6 hours into the event precedes the applicable time to reflux (15.6 hours) calculated in PWROG-14027-P by 4 hours. However, the NRC staff noted that, in addition to the issues discussed above, the time to reflux determined in PWROG-14027-P did not consider the potential for increased leakage due to seal faceplate degradation that is expected to occur in an ELAP event due to the loss-of-seal cooling. This hydrothermal corrosion phenomenon is discussed in greater detail in the following section of this SE. The NRC staff performed confirmatory simulations to determine the expected combined impact of the hydrothermal corrosion phenomenon along with the differences in initial RCS volume and accumulator injection. The NRC staff's confirmatory simulations concluded that Unit 2 could be expected to enter reflux cooling at 13.6 hours into the event, in the absence of FLEX RCS injection. On the basis of these simulations, the NRC staff concluded that, although the available margin may be reduced relative to the licensee's determination, providing RCS makeup by 11.6 hours into the event would be capable of maintaining adequate natural circulation flow in the RCS for CNS, Units 1 and 2.

Therefore, based on the evaluation above, the licensee's analytical approach should appropriately determine the sequence of events for reactor core cooling, including time-sensitive operator actions, and evaluate the required equipment to mitigate the analyzed ELAP event, including pump sizing and cooling water capacity.

3.2.3.3 Reactor Coolant Pump (RCP) Seals

Leakage from RCP seals is among the most significant factors in determining the duration that a PWR can cope with an ELAP event prior to initiating RCS makeup. An ELAP event would interrupt cooling to the RCP seals, resulting in the potential for increased seal leakage and the 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.

All four Model 93A RCPs installed at each CNS unit use standard three-stage Westinghouse seal packages. Per the discussion in Section 3.2.3.2, the licensee is relying on thermal-hydraulic analysis performed with the NOTRUMP code, as documented in WCAP-17601-P and PWROG-14064-P, to determine the time at which makeup would be required to maintain adequate natural circulation flow in the RCS. In accordance with analysis and testing documented in WCAP-10541-P, Revision 2, "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," the ELAP analysis in WCAP-17601-P assumed a leakage rate at nominal post-trip cold leg conditions (i.e., 2250 pounds per square inch absolute (psia) and 550°F) of 21 gpm for each of the four RCPs, plus an additional 1 gpm of operational leakage. In the WCAP-17601-P 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.

Subsequent 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. These concerns are documented in Westinghouse Nuclear Safety Advisory Letter (NSAL) 14-1, dated February 10, 2014, including: (1) the initial post-trip leakage rate of 21 gpm does not apply to all Westinghouse pressurized-water reactors 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. To address these issues, the PWROG performed additional analytical calculations using Westinghouse's seal leakage model (i.e., ITCHSEAL). These calculations included: (1) benchmarking calculations against data from RCP seal leakage testing, 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, the licensee indicated that CNS is relying on the generic Westinghouse RCP seal leakage calculations that have been performed by the PWROG. The generic PWROG calculations audited by the staff, including proprietary reports PWROG-14015, "No. 1 Seal Flow Rate for Westinghouse Reactor Coolant Pumps Following Loss of All AC Power," and PWROG-14027, classify CNS in the first generic analysis category (i.e., Category 1) specified in NSAL 14-1. 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. By letter dated February 15, 2016 [Reference 19], the licensee confirmed the Category 1 status of the seal leakoff line. During the audit, the NRC staff reviewed configuration information provided by the licensee to confirm that the leakoff line hydraulic characteristics for CNS are bounded by the assumed characteristics analyzed for Category 1.

To ensure that the generic Category 1 leakage rates are applicable to CNS, the NRC staff requested during the audit that the licensee confirm that applicable portions of the first-stage seal leakoff line piping can withstand the maximum pressure expected during an ELAP event. According to generic calculations performed by Westinghouse using the ITCHSEAL code, Category 1 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 initial RCS cooldown. Therefore, to support application of the generic Category 1 leakage rates, it is necessary to demonstrate that a rupture in the pressure boundary of leakoff line piping or components upstream and inclusive of the flow orifice would not occur at CNS. By letter dated February 15, 2016, the licensee informed the NRC staff that the applicable portions of the leakoff line piping and components had been evaluated and determined to be adequately protected from a pressure transient up to 2500 psia. The licensee also stated that thicker orifice plates have been installed in the first-stage seal leakoff lines to ensure integrity of this pressure boundary under analyzed ELAP conditions. Thus, the licensee's analysis concluded that the functionality of the first-stage seal leakoff lines should not be challenged during an analyzed ELAP event and Category 1 leakage rates should be applicable to CNS.

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 data from WCAP-10541-P that documents an RCP seal leakage test 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 with respect to RCS 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 was brought to the staff's attention. The leakage rate during this test was significantly higher than that of the test in WCAP-10541-P 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 staff determined should offset the potential for increased leakage rates observed in the additional Montereau test.

In conjunction with the revised seal leakage analysis that Westinghouse performed for the first-stage seal, 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, additional leakage past the second-stage seal could add to the first-stage seal leakoff line flow that has been considered in the licensee's evaluation. Previous calculations documented in WCAP-10541-P 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 the 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 as part of AREVA's seal development program (discussed further below) have indicated that the second-stage seals could open and remain open under ELAP conditions. This unexpected phenomenon occurred near the end of the tests and could not be fully understood and evaluated by the vendors or NRC staff, based upon the limited data available. While considering these limitations, the staff observed that the opening of the second-stage seal did not appear to result in an increase in the total rate of leakage measured during the two AREVA tests.

On March 3, 2015, Westinghouse issued Technical Bulletin (TB) 15-1, "Reactor Coolant System Temperature and Pressure Limits for the No. 2 Reactor Coolant Pump Seal." Through TB 15-1, Westinghouse communicated to affected customers that long-term integrity of Westinghouse-designed second-stage RCP seals could not be supported by the available analysis, and recommended that affected plants execute an extended cooldown of the RCS to less than 350°F and 400 psig by 24 hours into the ELAP event. Second-stage seal integrity appears necessary to ensure that leakage from Westinghouse-designed RCP seals can be limited to a rate that can be offset by the FLEX equipment typically available for RCS injection under ELAP conditions. The NRC is aware that a revised TB 15-1 has been issued to affected licensees, including CNS, as of August 29, 2016. Although the staff has not performed an in-depth review of the adequacy of the revised TB 15-1 cooldown profiles, the staff notes that CNS appears to comply with the Westinghouse recommendations. The staff also expects that licensees will evaluate new information relevant to plant structures, systems and components, (such as Westinghouse TBs) for action, as appropriate, in accordance with the site corrective action program (CAP).

The seal leakoff analysis discussed above assumes no failure of the seal design, including the elastomeric O-rings. During the audit review, the licensee confirmed that all installed RCP seal O-rings at CNS are the high-temperature-qualified 7228-C type, and that only equivalent or better O-rings will be used in the future. Therefore, the staff's audit review concluded that O-ring failure for CNS under analyzed ELAP event conditions would not be expected.

During the audit review, the licensee confirmed that, following the loss of seal cooling that results from the ELAP event, seal cooling would 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 increased seal leakage.

In addition, the NRC staff audited information associated with the more recent RCP seal leakage testing performed by AREVA. The AREVA testing showed a gradual increase in the measured first-stage seal leakage rate, which post-test inspection and analysis tied to hydrothermal corrosion of silicon nitride (likely assisted by flow erosion). Silicon nitride ceramic is used to fabricate the first-stage seal faceplates currently in operation in Westinghouse-designed RCP seals. This material degradation phenomenon would not have been 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). However, hydrothermal corrosion of silicon nitride became an audit focus area because the test data indicated that the long-term seal leakage rate could exceed the values assumed in the licensee's analysis. Academic research reviewed by the industry and NRC staff associated with this general phenomenon indicated that the corrosion rate is temperature dependent.

From the limited information available regarding the recent AREVA tests, as well as several sensitivity calculations performed by the NRC staff 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/or 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. The licensee's FIP states that initiating RCS makeup at 11.6 hours would ensure that natural circulation can be maintained in the RCS. Catawba's initial RCS makeup flow capacity of 40 gpm, which can be augmented to 60 gpm using the injection pump's diesel driver, exceeds the total rate of RCS leakage predicted by the PWROG's analysis following RCS depressurization, such that RCS inventory would begin to recover upon restoration of RCS makeup. However, in light of the potential for hydrothermal corrosion behavior, such as observed during the AREVA testing, the NRC staff determined that the mitigating strategy documented in the licensee's FIP could allow the RCP seal leakage rate to increase with time, potentially to the point of exceeding the available FLEX injection capacity. Due to the temperature-dependence of the hydrothermal corrosion reaction discussed above, the time and target temperature of the RCS cooldown have a significant impact on the long-term leakage rate from the RCP seals. At the time of developing the mitigating strategy documented in its FIP, the licensee was not fully aware of the influence of hydrothermal corrosion on the long-term seal leakage rate and had not specifically analyzed the potential impacts.

Therefore, during the audit, the NRC staff performed confirmatory evaluations that used empirical hydrothermal corrosion data to estimate the expected impact of faceplate degradation on the RCP seal leakage rate during the analyzed ELAP event. According to the staff's calculations based on the cooldown profile described in the FIP, hydrothermal corrosion could result in greater RCS leakage for CNS than had been considered in the licensee's thermal-hydraulic analysis. To address this concern, by letter dated September 27, 2016, the licensee provided supplementary cooldown projections for the postulated ELAP. Considering the information in the FIP, as supplemented, the approximate cooldown profile for CNS in the analyzed ELAP event is summarized as follows:

Elapsed Time (hrs)	RCS Cold Leg Temperature (° F)
0	561
2.6	403
25	371
37	< 350

The NRC staff subsequently revised its confirmatory evaluations to reflect the above cooldown profile. Based upon the current understanding of the hydrothermal corrosion phenomenon, the revised staff evaluations determined that reducing the RCS cold leg temperature below 350°F by 37 hours into the event should be sufficient to terminate the hydrothermal corrosion reaction prior to the RCP seal leakage rate increasing beyond the long-term capability of the CNS's FLEX RCS makeup strategy. As a result, the NRC staff concluded that the FLEX RCS makeup strategy for CNS should be capable of satisfying the requirement in Order EA-12-049 for indefinite coping.

Based upon the preceding discussion, the NRC staff concludes that the RCP seal leakage rates assumed in the licensee's thermal-hydraulic analysis, as modified in the evaluation above to account for the impacts of hydrothermal corrosion, may be applied to the beyond-design-basis ELAP event for the site.

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 sub-criticality 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, which (according to the core operating history assumed in NEI 12-06) would
    - initially increase above its equilibrium value following reactor trip, thereby adding negative reactivity
    - peak at roughly 12 hours post-trip and subsequently decay away gradually, thereby adding positive reactivity
- the passive injection of borated makeup from nitrogen-pressurized 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 and are determined by plant-specific analysis.

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, 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 staff expects that any core design changes, such as those considered in a core reload analysis, will be evaluated to determine that they do not adversely impact the approved FLEX strategies, especially the analyses which demonstrate that re-criticality will not occur during a FLEX RCS cooldown.

During the audit, the NRC staff reviewed the licensee's shutdown margin calculation. According to the FIP, borated water from the FWST will be injected into the RCS no later than 11.6 hours into the event. The licensee's shutdown margin analysis conservatively determined that the injection of borated coolant should begin by 13.85 hours into the event to ensure that re-criticality can be avoided as the core's xenon concentration decays away. The calculation of the time for initiating RCS boration was based upon the conservative assumption of an initial RCS temperature of 350°F. The licensee further determined the required rate of injection to maintain the reactor subcritical by considering the boration rate necessary to counterbalance the rate of reactivity increase due to xenon decay. The licensee calculated that a FLEX RCS makeup rate of 40 gpm, in conjunction with RCS letdown via the reactor vessel head vent, would provide sufficient capacity to maintain the reactor subcritical. The licensee's analysis considered several cases, which varied the assumed values for accumulator injection volume and RCS leakage. The licensee further calculated that 28,100 gallons of FWST water (for both units combined) which is borated to a minimum concentration of 2700 parts per million (ppm), is required to ensure long-term sub-criticality at an RCS temperature of 350°F, and administrative controls ensure that this volume will remain valid for future core designs. This volume is well within the approximately 105,000-gallon volume that the licensee calculated would remain in each of the two FWSTs, even assuming the analyzed worst-case missile damage.

Toward the end of an operating cycle, when the RCS boron concentration reaches its minimum value, some PWR licensees may need to vent the RCS to ensure that their FLEX strategies can inject a volume of borated coolant that is sufficient to satisfy shutdown margin requirements in cases where minimal RCS leakage occurs. During the audit, the NRC staff reviewed CNS's capability to conduct RCS venting in the case that letdown from the RCS is necessary. In this case, operators would follow the direction of FSG-8, "Alternate NC [RCS] System Boration," Revision 1. The applicable steps of this FSG would energize and open one of two pairs of dc-powered reactor vessel upper head vent valves from the control room. The head vent path would be opened in response to high pressurizer pressure or level, and closed again on indication of low pressurizer level or when operators have completed the required boron addition.

Attachment 7 of the licensee's compliance letter notes several key aspects of its shutdown margin analysis that satisfy and in some cases conservatively surpass endorsed regulatory guidance, including:

- Zero RCS leakage through the RCP seals is assumed, which maximizes the required boron injection.

- Limiting core conditions are assumed with regard to power history, time-in-life, initial RCS boron concentration, and xenon concentration. The calculation assumes a 500-day effective full power day run at core end-of-cycle (EOC) prior to the ELAP event. Initial RCS boron concentration is assumed to be 6 ppm, consistent with conditions at full power and EOC. Initial core xenon concentration is assumed to be at equilibrium.

- The assumed required final boron concentration in the RCS is set conservatively high (i.e., at 475 ppm), increasing the required injection volume.

- An hour is subtracted from the required response time, to reflect a one-hour delay to support adequate boron mixing throughout the RCS volume.

- The time requirement to begin RCS injection is based on the post-cooldown target temperature of 350°F, whereas, according to existing procedures, operators would actually hold the RCS temperature at 420°F until boration is completed.

The NRC staff made several further observations concerning the licensee's shutdown margin calculation. First, the assumed RCS injection rate from the high pressure FLEX pump was limited to 40 gpm; whereas, the licensee states that the pump's flow rate can actually be increased by approximately 50 percent during RCS depressurization via the use of its diesel driver. In addition, the staff observed that the licensee's assumption of the letdown flowrate achievable through the reactor vessel upper head vent path was based on the conservative assumption that the fluid in the reactor vessel upper head would remain saturated at a pressure of 300 psia throughout the boration evolution. Whereas, in a case corresponding to zero RCS leakage, the actual pressure during an ELAP event should initially exceed this value; and further, with no RCS leakage, refilling the vessel head and restoring pressurizer level should result in RCS pressurization and sub-cooling of the fluid in the vessel head, both of which would further increase the achievable letdown flowrate. This assumption is particularly significant inasmuch as the licensee calculated that, in the limiting case, a large part of the required boration time would be allocated to venting the RCS. Based on its audit review, the NRC staff agreed with the licensee's conservative determination that boration to support shutdown margin requirements at 350°F could be completed by 24.5 hours into the ELAP event for CNS, Unit 2 and by 37.1 hours into the event for CNS, Unit 1; the staff further expected that the required boron concentration could realistically be achieved well in advance of these times. Subsequent to CNS's onsite audit, the licensee performed additional analysis that included a realistic estimate of fluid sub-cooling in determining the choked flowrate through the RCS vent flow path. Based on this refined analysis, the licensee determined that the required boration for CNS, Unit 1 could be completed within 24 hours of event initiation.

The NRC staff's audit review of the licensee's shutdown margin calculation further determined that credit was taken for uniform mixing of boric acid during the ELAP event. The NRC staff had previously 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 PWROG submitted a position paper, dated August 15, 2013 (withheld from public disclosure due to proprietary content), which 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 41], 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.

By letter dated March 31, 2016 [Reference 40], the licensee confirmed that CNS had addressed the concerns in the NRC endorsement of the PWROG position paper on boron mixing in the site boration analysis. The NRC staff verified that the licensee's analyses considered the appropriate range of RCS leakage rates. The NRC staff further confirmed that the licensee would provide RCS makeup prior to RCS loop flow decreasing below the single-phase natural circulation flow rate. Finally, the NRC staff also confirmed that the licensee's calculations adequately incorporated a 1-hour delay to account for the mixing time of boric acid in the RCS.

The NRC staff's audit also identified that, while the licensee's analyses explicitly demonstrate adequate shutdown margin for an RCS average temperature of 350°F or greater, within 37 hours of event initiation, the licensee would reduce the RCS temperature beyond this point to mitigate the impacts of hydrothermal corrosion to the RCP seals. Maintaining an SG pressure of 80-100 psig, as described above, indicates that the RCS cold leg temperature could be reduced to approximately 324°F. In addition, within 6 days of event initiation, the licensee expects to further cool the RCS to Mode 5 conditions (i.e., RCS average temperature ≤ 200°F). As a result, maintaining adequate shutdown margin would require the injection of additional boron into the RCS prior to reducing RCS average temperature below 350°F. Although the licensee's conclusion that adequate capability exists to supply the additional boration required by 37 hours (and again at 6 days) into the event appears reasonable in light of the capabilities discussed above for CNS, at the time of the NRC staff's audit, the reactivity impacts and required boration times for these additional cooldowns had not been explicitly analyzed by the licensee and thus could not be specifically reviewed by the staff. Therefore, the NRC staff performed a confirmatory evaluation to assess the impact of varying the completion time for borating the RCS sufficiently to support depressurizing the SGs to 80-100 psig. The NRC staff's evaluation determined that, for the analyzed ELAP event, the licensee's planned completion time of 37 hours should provide ample margin to the time at which this cooldown must be completed to limit RCP seal leakage to within the capacity of the FLEX RCS makeup strategy.

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

3.2.3.5 FLEX Pumps and Water Supplies

For SG makeup, the licensee described in the FIP a FLEX low pressure pump (diesel-powered high volume pumps) with the function of supplying water from the SNSWP to the FLEX raw water distribution system after the operators secure the TDAFW pumps. The FLEX Storage Facility (FSF) will contain two FLEX low pressure pumps, since one pump is designed to supply both units. This would satisfy the "N+1" provision in NEI 12-06. The licensee also described the FLEX medium pressure pumps (diesel-powered, centrifugal) that are used to provide SG makeup (and RCS makeup in Modes 5 or 6). The pumps are connected in series with hoses connected to the FLEX low pressure pump to obtain SNSWP water and supply the SGs at around 24 hours after ELAP is declared. The licensee indicated that CNS has three portable FLEX medium pressure pumps (one per unit and one spare pump) stored in the FSF to satisfy the "N+1" inventory provision. For RCS makeup, the licensee described in its FIP, a FLEX high pressure pump (diesel-powered, centrifugal pump) to provide RCS system injection from the FWST. The licensee also stated that CNS has three portable FLEX high pressure pumps, stored in the FSF, to satisfy the "N+1" inventory provision. The water supplies for all three pumps to be used for SG and RCS makeup are described with more detail in Section 3.10 of this SE. The licensee also described FLEX portable sump pumps to mitigate potential internal flooding of areas near the TDAFW pump pits and the Auxiliary Building. The licensee stated that CNS has six submersible FLEX sump pumps (three for TDAFW pump pits and three for the Auxiliary Building), which are all stored and deployed from the FSF.

Section 11.2 of NEI 12-06 states that design requirements and supporting analysis should be developed for portable equipment that directly performs a FLEX mitigation strategy for core, containment, and SFP that provides the inputs, assumptions, and documented analysis that the mitigation strategy and support equipment will perform as intended. During the audit review, the licensee provided for the staff's review FLEX hydraulic calculations (CNC-1223.02-00-0025, "Flow Model SNSWP to CA Connections for Phase 2 FLEX Strategies," Revision 0, and CNC-1223.02-00-0026, "Flow Model of SNSWP to RN Connections and CA for Phase 2 Flex Strategies," Revision 0), which both evaluated the use of the FLEX pumps receiving makeup water from the SNSWP to supply the SGs and distributing through the auxiliary feedwater and nuclear service water systems respectively. The licensee also provided for the staff's review FLEX hydraulic calculation CNC-1223.02-00-0028, "Flow Model for U1/U2 NI Portable Pump Injection to RCS Phase 2 & 3 FLEX Strategies," Revision 1, which evaluated the use of FLEX pumps providing makeup to the RCS from the FWST. The staff was able to confirm that flow rates and pressures evaluated in the hydraulic analyses were reflected in the FIP for the respective SG and RCS makeup strategies based upon the above FLEX pumps being deployed and implemented as described in the FSGs. The staff also conducted a walk down of the hose deployment routes for the above FLEX pumps during the audit to confirm the evaluations of the hose distance runs in the above hydraulic analyses.

Based on the staff's review of the FLEX pumping capabilities at CNS, as described in the above hydraulic analyses and the FIP, the licensee has demonstrated that its portable FLEX pumps should perform as intended to support core cooling and RCS inventory control during an ELAP caused by a BDBEE, consistent with NEI 12-06, Section 11.2.

3.2.3.6 Electrical Analyses

The licensee's electrical strategies provide power to the equipment and instrumentation used to mitigate the ELAP and loss of normal access to the UHS. The licensee's strategy for RCS inventory control uses the same electrical strategy as 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. The NRC staff reviewed the licensee's FIP, conceptual electrical single-line diagrams, summaries of calculations for sizing the FLEX diesel and turbine generators and station batteries, and summaries of calculations that addressed the effects of temperature on the electrical equipment as a result of loss of heating, ventilation, and air conditioning (HVAC) during the postulated ELAP.

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

During the first phase of the ELAP event, CNS would rely on the station's safety-related batteries to provide power to instrumentation for monitoring parameters and power to controls for SSCs used to maintain the key safety functions (core cooling, RCS inventory control, and Containment integrity). The CNS station batteries and associated dc distribution systems are located within the Auxiliary Building, which is a Seismic Category I structure. The batteries are therefore protected from the applicable extreme external hazards. During the audit review the NRC staff noted that the licensee's procedures (ECA-0.0, "Loss of All AC Power," Revision 49 for Unit 1 and Revision 48 for Unit 2), direct operators to conserve dc power during the event by stripping nonessential loads. The plant operators would commence stripping, or load shedding, dc loads within 2.5 hours after the occurrence of an ELAP. The licensee expects load shedding to be completed within 1 hour.

Each of the units at CNS are provided with four 125 Volts-dc (Vdc) safety-related batteries (EBA, EBB, EBC, and EBD). The safety-related batteries are arranged as A Train (EBA, EBC) and B Train (EBB, EBD). The safety-related batteries are GNB NCN-21 that are rated for 1495 ampere-hours at an 8 hour discharge rate to 1.75 V per cell. The licensee noted and the staff confirmed that the useable station battery capacity could be extended up to 11.5 hours for the limiting battery (EBA) by load shedding non-essential loads.

In its FIP, the licensee noted that it had followed the guidance in an NEI white paper dated August 27, 2013, regarding extended battery duty cycles [Reference 42] when evaluating the batteries under ELAP conditions. This paper was endorsed by the NRC by letter dated September 16, 2013 [Reference 43]. To further confirm extended battery duty cycles, the NRC sponsored testing at Brookhaven National Laboratory that resulted in the issuance of NUREG/CR-7188, [Reference 56] in May of 2015. The purpose of this testing was to examine whether existing vented load acid batteries can function beyond their defined design-basis (or beyond-design-basis if existing station blackout 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 dc power to the core-cooling equipment 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 battery manufacturing performance data.

The NRC staff reviewed the dc coping study, CNC 1381.05-00-0122, "Station Blackout Battery Sizing Calculation for the 125 VDC Vital I&C Batteries," Revision 6, which verified the capability of the dc system to supply the required loads during the first phase of the CNS FLEX mitigation strategy plan for an ELAP as a result of a BDBEE. The licensee's evaluation identified the required loads and their associated ratings (ampere and minimum required voltage) and the loads that would be shed within 3.5 hours to ensure battery operation for least 11.5 hours. This provides ample margin to transition to Phase 2 as the licensee expects Phase 2 equipment to be deployed within 10 hours of the onset of an ELAP as a result of a BDBEE.

Based on the information contained in NUREG/CR-7188, the staff's review of the licensee's analysis, the battery vendor's capacity and discharge rates for the batteries, and the licensee's procedures, the NRC staff finds that the CNS, Units 1 and 2, dc systems should have adequate capacity and capability to supply power to required loads until the battery chargers are energized by the FLEX DGs, provided that the necessary load shedding is completed within the times analyzed in the licensee's calculations.

The licensee's Phase 2 strategy includes repowering 600 Vac buses within 11.5 hours using a portable 500-kilowatt (kW) 600 Vac FLEX DG (one per unit). There are three portable FLEX 600 Vac diesel generators to satisfy the "N+1" provision of NEI 12-06. The licensee's transition to Phase 2 is expected to occur earlier than the calculated depletion of the Class 1E 125 Vdc batteries (within 9 hours versus a calculated battery capacity of 11.5 hours). The portable 600 Vac FLEX DGs would supply power to CNS, Units 1 and 2 vital 600 Vac vital bus circuits providing continuity of key parameter monitoring and other required loads.

The NRC staff reviewed engineering change (EC) evaluation EC401541, "FLEX Diesel Generator Loading Evaluation," Revision 0, and procedures FSG-20, "Flex Electrical Distribution", Revision 3, and FSG-05, "Initial Assessment and FLEX Equipment Staging," Revision 4, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the Class 1E equipment. Since the loads for Unit 1 and 2 are almost identical, and Unit 2 will power more loads, the licensee only evaluated the required loads for Unit 2. Based on the NRC staff's review, the minimum required loads for Phase 2 equate to 422 kW for Unit 2. Therefore, one 500 kW FLEX DG per unit is adequate to support the electrical loads required for Phase 2 strategies. Furthermore, the licensee's Phase 2 electrical strategy ensures that the safety-related battery chargers will be energized prior to the batteries depleting below minimum acceptable voltage.

For Phase 3, the licensee plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources. The offsite resources that will be provided by the NSRC includes four (2 per unit) 1-MW 4160 Vac combustion turbine generators, two (1 per unit) 1100 kW 480 Vac combustion turbine generators, two (1 per unit) 480 Vac to 600 Vac transformers, and distribution panels (including cables and connectors). Each portable 4160 Vac combustion turbine generator is capable of supplying approximately 1 MW, but two combustion turbine generators will be operated in parallel to provide approximately 2 MW. Based on EC evaluation EC401541, the total load for Phase 3 is 979 kW. The 480 Vac combustion turbine generator has a capacity of 1100 kW and would primarily be used to power the LCVUs to assist in long-term containment temperature control. Sufficient margin exists between the calculated loading and the capacity of the 4160 Vac and 480 Vac turbine generators being supplied by the NSRC to ensure that the minimum required loads can function as expected. Based on its review, the NRC staff finds that the 4160 Vac and 480 Vac equipment being supplied from the NSRCs has sufficient capacity and capability to supply the required loads.

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

3.2.4 Conclusions

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

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-2 and Appendix D, summarize an acceptable approach consisting of three separate capabilities for the SFP cooling strategies. This approach uses a portable injection source to provide the capability for 1) makeup via hoses on the refueling floor capable of exceeding the boil-off rate for the design-basis heat load; 2) makeup via connection to SFP cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load; and 3) spray via portable monitor nozzles from the refueling floor using a portable pump capable of providing a minimum of 200 gpm per unit (250 gpm if overspray occurs). During the event, the licensee selects the method to use based on plant conditions. This approach also requires a strategy to mitigate the effects of steam from the SFP, such as venting.

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

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

The ELAP causes a loss of cooling in the SFP. As a result, the pool water will heat up and eventually boil off. The licensee's response is to provide makeup water and a ventilation pathway. The timing of operator actions and the required makeup rates depend on the decay heat level of the fuel assemblies in the SFP.

3.3.1 Phase 1

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

3.3.2 Phase 2

The licensee stated in its FIP that SFP makeup will be provided from the SNSWP using the FLEX low pressure pump to compensate for SFP boil-off. The primary SFP makeup strategy pressurizes the nuclear service water system and then opens an installed nuclear service water system valve and an installed SFP cooling system valve to establish a flow path to the SFP using installed piping. The alternate makeup strategy is described in the FIP as pressurizing the nuclear service water system and using a hose connection from the nuclear service water piping to a valve in the SFP cooling system skimmer loop. The licensee stated that operators will attach an adapter to this valve to facilitate the hose connection. Catawba does not incorporate a FLEX strategy for spray makeup to the SFP, contrary to the provisions of NEI 12-06, Revision 0. This alternative to the guidance is further discussed and evaluated by the NRC staff in Section 3.14.2 of this SE.

3.3.3 Phase 3

The licensee stated in its FIP that the portable combustion turbine generator from the NRSC will be used to re-power the installed SFP cooling and component cooling water pumps to maintain SFP cooling long-term. The cooling of the component cooling water system will be from the nuclear service water system that is being supplied from the portable diesel-driven pump located at the SNSWP. The licensee also stated that additional diesel fuel for portable equipment will be brought in from off-site resources when required.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

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

The licensee credited the SFP Building, the Auxiliary Building, and the Nuclear Service Water Pumphouse in the FIP as the plant SSCs needed for SFP cooling. The three buildings are described as Seismic Category I structures that are designed to provide protection from all external hazards. The licensee also stated that the SFP cooling system, component cooling water system, and nuclear service water systems are also used as part of the SFP makeup strategy and are all robust systems that are protected from all of the external hazards.

During the audit review, the licensee provided NAI-1813-002, "Catawba Fuel Handling Building Extended Loss of AC Power FLEX Response," Revision 0, for the staff to review the licensee's assessment of habitability on the SFP refuel floor. This document and the FIP indicate that boiling begins at approximately 37 hours during a normal, non-outage situation.

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. The operators are directed to open selected doors in the SFP Building within 6 hours after an ELAP is declared to minimize the impact of condensed steam on Auxiliary Building habitability. The licensee also stated that the Auxiliary Building doors will be secured and door cracks will be sealed to further prevent condensed steam from entering other parts of the Auxiliary Building outside of the selected vent path.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves use of the FLEX low pressure pump with associated hoses and fittings taking suction from the SNSWP to supply makeup water to the SFP. The staff's evaluation of the connection points for the FLEX pump is discussed in Section 3.7.3.1 of this SE and the staff's evaluation of the SNSWP is discussed in Section 3.10.3 of this SE.

3.3.4.1.2 Plant Instrumentation

In its FIP, the licensee stated that the instrumentation for SFP level will meet the requirements of Order EA-12-051. The NRC staff's review of the SFP level instrumentation, including the primary and backup 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

Section 11.2 of NEI 12-06 states, in part, that design requirements and supporting analysis should be developed for portable equipment that directly performs a FLEX mitigation strategy for core, containment, and SFP that provides the inputs, assumptions, and documented analysis that the mitigation strategy and support equipment will perform as intended. In addition, NEI 12-06, Section 3.2.1.6, Condition 4 states that SFP heat load assumes the maximum design basis heat load for the site. In accordance with NEI 12-06, the licensee performed a thermal-hydraulic analysis of the SFP as a basis for the inputs and assumption used in its FLEX equipment design requirements analysis. During the audit, the licensee provided calculation DPC-1201.30-00-0012, "Spent Fuel Pool Loss of Cooling Heat Up Time Determination Due to Decay Heat," Revision 3, for the staff's review of the thermal-hydraulic analysis for the CNS SFP. The licensee evaluated the SFP with the worst case heat load corresponding to a full core off load 6 days after a shutdown will produce evaporation rates of 96.8 gpm per SFP. The evaluation also concluded that the SFP will begin to boil at 8.8 hours after an ELAP/loss of normal access to the UHS event in the worst case scenario with the assumption of a maximum starting SFP temperature of 125°F and a full core offload during an outage after 6 days. For normal conditions of initial pool temperature of 125°F immediately following a typical 21-day refueling outage, the licensee concluded that the minimum time to boil is extended to about 37 hours. The staff evaluated the licensee's document during the audit to verify that the licensee's analyses of utilizing the FLEX low pressure pumps were capable of providing the necessary flow needed for SFP makeup.

Also, during the audit review, the NRC staff identified that the Standby Shutdown Facility (SSF) for CNS would be available in some ELAP scenarios, and as described in procedure ECA-0.0, it could preferentially be used to mitigate an ELAP event. Specifically, if available, the SSF equipment would be used to pump coolant from the SFP into the RCS through the RCP seal injection flow path. The licensee's FIP did not assess the overall impact on the SFP level in such a scenario where the SFP inventory is being depleted by both RCP seal makeup and evaporation. By letter dated September 27, 2016 [Reference 58], the licensee supplemented the FIP with an evaluation of the SFP makeup time with the SSF equipment in use to support ELAP strategies. The licensee stated that SFP makeup would be required by 71 hours in a normal scenario after an ELAP is declared with the SSF in service to maintain SFP level 10 feet above the fuel racks. The NRC staff evaluated calculation DPC-1201.30-00-0012, the supplemental information provided by the licensee, and the projected flow rates to confirm that the licensee's SFP makeup strategy would not be adversely impacted if the SSF is utilized to supply RCP seal injection water after an ELAP.

Based on the evaluation above, the NRC staff finds that the licensee appears to have provided a sufficient analysis that considered maximum design-basis SFP heat load during operating, pre-fuel transfer or post-fuel transfer operations, the basis for assumptions and inputs used in determining the design requirements for FLEX equipment used in SFP cooling consistent with NEI 12-06 Section 3.2.1.6, Condition 4 and Section 11.2.

3.3.4.3 FLEX Pumps and Water Supplies

As described in the FIP, the SFP cooling strategy relies on the FLEX low pressure pump to provide SFP makeup during Phase 2. In the FIP, Section 2.3.7 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX low pressure pump. The staff noted that the performance criteria of a FLEX pump supplied from an NSRC for Phase 3 would allow the NSRC pump to fulfill the mission if the onsite FLEX low pressure pump were to fail. As stated in the FIP, the SFP makeup rate capability meets or exceeds the most conservative steaming rate.

3.3.4.4 Electrical Analyses

The staff performed a comprehensive analysis of the licensee's electrical strategies, which includes the SFP cooling strategy. The electrical components credited by the licensee as part of its FLEX mitigation strategies, outside of instrumentation to monitor SFP level, are the NSRC 4160 Vac combustion turbine generators used to repower the installed SFP cooling and component cooling water pumps to provide long-term cooling via normal means. As described in Section 3.2.3.6 of this SE, the staff reviewed the licensee's electrical evaluation and determined that the portable FLEX combustion turbine generators should have sufficient capacity and capability to supply these loads, if necessary.

3.3.5 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that if implemented appropriately should maintain or restore SFP cooling following an ELAP consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order. As stated above, the licensee has provided an alternate method to exclude the spray makeup capability. The justification and NRC staff's evaluation of this alternate method is discussed in SE Section 3.14.2.

3.4 Containment Function Strategies

The industry guidance document, NEI 12-06, 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. The CNS units each have an ice condenser containment.

The licensee performed a containment evaluation, DPC-1552.08-00-0280, "Extended Loss of AC Power (ELAP) - Ice Condenser Containment Response with FLEX Mitigation Strategies," Revision 2, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation (Case 2) analyzed the strategy of repowering a hydrogen skimmer fan 24 hours after an ELAP-inducing event, repowering two LCVUs 48 hours following an ELAP-inducing event, and repowering a containment air return fan 52 hours after the ELAP-inducing event. During the audit, the NRC staff reviewed a description and procedural steps to accomplish these tasks in CNS's FSG-12, "Alternate Containment Cooling," Revision 1.

The calculation concludes that the containment temperature remains well below the Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.1.2 design limit of 250°F for at least 7 days when this strategy is implemented. The UFSAR Section 6.2.1.1.2 design limit of 15 psig is shown to be exceeded by approximately 2.38 psig (Unit 1) and 2.5 psig (Unit 2) for a 12 hour period in the Case 2 analytical model. A justification for this exceedance is provided in Section 3.4.4.2.

Additionally, although core damage is not expected, NEI 12-06, Table 3-2, guides licensees with ice condenser containments to repower the unit's hydrogen igniters by using a portable power supply as a defense-in-depth measure to maintain containment integrity. The CNS FIP states that one train of hydrogen igniters will be repowered in Phase 2 using a back-feed portable power strategy with the FLEX diesel generators.

3.4.1 Phase 1

The CNS containment analysis concludes that, aside from manual containment isolation activities, there are no Phase 1 actions required, as the containment pressure and temperature remain below their design limits. Wide range pressure instrumentation will be available to monitor containment conditions. Catawba's procedure(s) ECA-0.0, "Loss of All AC Power," and FSG-12, address containment isolation for each unit.

3.4.2 Phase 2

The CNS FIP states that one 500 kW, 600 Vac FLEX DG will be deployed for each unit. The FLEX electrical distribution system and FLEX DGs will be in place to provide power to the hydrogen igniters. Providing power to the containment hydrogen igniter assemblies would prevent the buildup of hydrogen gas, even though core damage is not expected if the licensee's core cooling strategy is successful as postulated for the ELAP event.

The FIP further states that the FLEX electrical distribution system and the FLEX DGs will also be used to repower hydrogen skimmer fans to help limit the temperature increase in the SG and pressurizer enclosures within 24 hours of the start of the ELAP event. In Phase 2, FSG-12 provides the controlling guidance for containment cooling.

3.4.3 Phase 3

During Phase 3, the NSRC-supplied 480 Vac generators along with energizing the 600V MCCs, will provide power to the LCVUs to limit the temperature increase within the SG and pressurizer enclosures. The FIP further states that a containment air return fan will also be repowered to mix the colder air in the ice condenser with the rest of containment to ensure that pressure and temperature conditions remain acceptable. In Phase 3, FSG-12 continues to provide the controlling guidance for containment cooling.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

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

3.4.4.1.1 Plant SSCs

Sections 1.2.2.4 and 3.8.2.1 of the CNS UFSAR state that the ice condenser containment consists of a free-standing, welded steel structure with a vertical cylinder, hemispherical dome with a flat base which is enclosed within a separate, reinforced concrete Reactor Building forming an annulus between the two structures. It is capable of withstanding an internal design pressure of 15 psig, and, as shown in UFSAR Table 3-1, both the containment and the surrounding Reactor Building are Seismic Category I structures. Table 3-1 further shows that the Reactor Building has been designed to protect against tornado loads and missiles.

Section 2.5.6 in the FIP, Thermal-Hydraulic Analysis, indicates the GOTHIC model analyzed that the maximum pressure in containment will exceed the 15 psig design pressure for a period of 12 hours. The maximum containment pressure is calculated to be 17.5 psig. The Containment Vessel at CNS was initially pressure tested to 17.25 psig. According to the licensee's FIP it has been analyzed to withstand a maximum pressure of 72 psig.

During the audit, NRC staff requested information regarding the robustness of the LCVUs and associated MCCs/ductwork credited in the FLEX strategies to control temperature and pressure inside containment. The licensee responded by letter dated September 27, 2016 [Reference 58]. The licensee stated that the seismic capability of the LCVUs was evaluated by FLEX robustness reports, as well as by the Expedited Seismic Evaluation Process (ESEP). The evaluations determined the "A" and "D" LCVUs on CNS, Units 1 and 2, were seismically robust. The licensee's letter also stated that the associated MCCs are seismically qualified and that all ductwork inside containment, including that associated with the LCVUs, is designed to survive a seismic disturbance.

Table 3-4 of the UFSAR shows that the containment air return fans and hydrogen skimmer fans are Safety Class 2 and designed for the applicable hazards. Electrical components are Class 1E.

Section 6.2.5.7 of the UFSAR states that the hydrogen mitigation system (which contains the hydrogen igniters) is seismically mounted. The hydrogen mitigation system is housed within the Reactor Building which, as stated above, has also been designed to protect components against tornado loads and missiles.

The NRC staff concludes that, based on these UFSAR qualifications, the Reactor Building, Containment, the hydrogen skimmer fans, the containment air return fans, and the hydrogen igniters credited in the strategy are robust, as defined by NEI 12-06, and should be available following an ELAP-inducing event. Additionally, the staff finds that the evaluation for the LCVUs shows that they are robust, as defined by NEI 12-06, and should be available following an ELAP-inducing event.

3.4.4.1.2 Plant Instrumentation

In NEI 12-06, Table 3-2, specifies that containment pressure is a key containment parameter which should be monitored by repowering the appropriate instruments. The licensee's FIP states that control room instrumentation would be available due to the coping capability of the station batteries and associated inverters in Phase 1, or the portable DGs deployed in Phase 2. If no ac or dc power was available, the FIP states that key credited plant parameters, including containment pressure, would be available using alternate methods.

3.4.4.2 Thermal-Hydraulic Analyses

The licensee provided the staff with containment evaluation DPC-1552.08-00-0280, "Extended Loss of AC Power (ELAP) - Ice Condenser Containment Response with FLEX Mitigation Strategies," Revision 3, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation utilized the GOTHIC computer code, version 8.0, to model the containment's pressure and temperature response to an ELAP event. The staff noted that the calculation contained two near-term cases of interest in evaluating the behavior of the containment. Both of the cases analyzed a 7-day coping period following an ELAP-initiating event. The results show the containment response to an ELAP event is a relatively slow moving transient. As such, the doors to the ice condenser are modeled to remain closed until a containment air return fan is re-powered and provides the necessary differential pressure to open them.

The analytical cases consider the leakage from the RCP seals to contribute heat and mass to the containment atmosphere at an initial rate of 5 gpm/pump which then increase to a maximum of 21 gpm/pump within the first 13 minutes of the transient, consistent with the sequence of events specified in WCAP-17601-P, Revision 1.

Case 1 of the calculation modeled the behavior of the containment parameters with no mitigation actions being taken. Under these conditions, the temperature in the upper and lower containment remains below the 250°F design limit for at least 4 days following the ELAP-initiating event. The containment pressure, however, was calculated to rise to the 15 psig design limit approximately 2 days after the initiating event. The local temperature within the pressurizer enclosure was calculated in this unmitigated case to rise above 280°F within 25 hours of the initiating event and continue to increase to approximately 320°F after 48 hours. The licensee determined that this continuous, localized temperature rise required mitigation.

Therefore, case 2 modeled the strategy of repowering a hydrogen skimmer fan at 24 hours, repowering two LCVUs at 48 hours, and repowering a containment air return fan at 52 hours following an ELAP-inducing event as described in procedure FSG-12. This model showed that repowering the hydrogen skimmer fan at 24 hours limited the pressurizer enclosure temperature rise to 280°F at hour 24. Repowering the two LCVUs at 48 hours further reduced the local temperatures in both the SG and pressurizer enclosures. In case 2, the containment pressure was shown to reach and exceed its 15 psig design limit approximately 43 hours after the initiating event. The calculation predicts a peak pressure of 17.38 (Unit 1) and 17.50 (Unit 2) psig. The repowering of the two LCVUs slowed the rate of pressure increase; however, the greatest benefit to all containment and enclosure parameters was shown to be realized after the containment air return fan was turned on at 52 hours and the doors to the ice condenser were opened. Following this action, the containment pressure and temperature quickly decreased and were shown to remain at low values for approximately 32 more hours before beginning to rise again. Case 2 showed the total time elapsed from the ELAP-initiating event to the time when the containment design pressure was reached again after opening the ice condenser doors was nearly 6 days.

It is appropriate to note that the offsite resources from the NSRC are expected to arrive onsite at CNS within 24 hours of the ELAP-initiating event. Case 2 of the calculation demonstrates that once the containment air return fan is repowered and the ice condenser doors are opened, the containment parameters of pressure and temperature quickly decrease and remain low until the ice inventory is depleted (approximately 32 hours from the time of opening). Thus, the staff finds it is likely that the strategy could be employed earlier than the timeline shown in the calculation, if necessary.

3.4.4.3 <u>FLEX Pumps and Water Supplies</u>

For Phase 1, Phase 2, and Phase 3, with the unit operating within the boundary conditions of NEI 12-06, Section 2, the analysis demonstrates that there are no mitigation strategies for which FLEX pumps or water supplies are required to maintain containment pressure below the design limit of 15 psig for over 6 days.

3.4.4.4 <u>Electrical Analyses</u>

The licensee performed a containment evaluation based on the boundary conditions described in Section 2 of NEI 12-06 to determine the temperature and pressure increase in the containment vessels resulting from an ELAP as a result of a BDBEE. Based on the results of the evaluation, the licensee developed required actions to ensure maintenance of containment integrity and required instrumentation function. With an ELAP initiated, while either CNS Units are in Modes 1-4, containment cooling for that unit is also lost for an extended period of time. Structural integrity of the reactor containment building due to increasing containment pressure will not be challenged during the first few days of an ELAP/loss of normal access to the UHS event. However, with no cooling in the containment, temperatures in the containment are expected to rise and could reach a point where continued reliable operation of key instrumentation might be challenged. The licensee's evaluations have concluded that containment temperature and pressure will remain below containment design limits and that key parameter instruments subject to the containment environment will remain functional for a minimum of five days. Therefore, actions to reduce containment temperature and pressure and to ensure continued functionality of the key parameters will not be required immediately.

The licensee's Phase 1 coping strategy for containment involves initiating and verifying containment isolation per procedure ECA-0.0. These actions ensure containment isolation following an ELAP/loss of normal access to the UHS. Phase 1 also includes monitoring wide range containment pressure using installed equipment. In its FIP, the licensee stated that during Phase 1 following an ELAP/loss of normal access to the UHS, the containment is initially cooled by an ice condenser. Therefore, steam escaping the primary and/or secondary systems is cooled as it rises from lower containment through the ice condenser and into upper containment.

The licensee's Phase 2 coping strategy is to use the portable FLEX DGs to repower one train of hydrogen skimmer fans within 24 hours of the start of the ELAP, which help limit the temperature increase in the SG and Pressurizer compartments. The SG and pressurizer compartment temperature limits will be challenged before overall containment temperature limits. The abnormally high temperatures in these compartments could affect associated level indications due to reference leg flashing. The Phase 2 coping strategy also includes repowering hydrogen igniters to prevent the buildup of hydrogen in case the ELAP event degrades to core damage. One train of hydrogen igniters will be re-powered using the back feed portable power strategy and the opposite train of hydrogen igniters can be repowered as an alternate. As described in Section 3.2.3.6 of this SE, the staff reviewed the licensee's evaluation and determined that the portable FLEX 600 Vac DGs have sufficient capacity and capability to supply the required Phase 2 loads, including those required to maintain containment temperature and pressure, and to ensure that key instrumentation remains functional.

The licensee's Phase 3 coping strategy is to use the 4160 Vac combustion turbine generators provided by the NSRC to repower two LCVUs within 48 hours to limit the temperature increase within the SG and pressurizer compartments. Additionally, a CARF will be repowered within 52 hours to establish an air flow path through the ice condenser, reduce containment pressure, and limit the further heat-up of the SG and pressurizer compartments. Once the CARF is operating, SG and pressurizer compartment temperature will be maintained below 200°F and containment pressure will be maintained less than 6 psig. As described in Section 3.2.3.6 of this SE, the staff reviewed the licensee's evaluation, and determined that the portable FLEX 4160 Vac and 480 Vac combustion turbine generators have sufficient capacity and capability to supply the required Phase 3 loads, including those required to reduce containment temperature and pressure, and to ensure that key instrumentation remains functional.

3.4.5 Conclusions

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

3.5 Characterization of External Hazards

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

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

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

References to external hazards within the licensee's mitigating strategies and this SE are consistent with the guidance in NEI-12-06 and the related NRC endorsement of NEI 12-06 in JLD-ISG-2012-01. Guidance document NEI 12-06 directed licensees to proceed with evaluating external hazards based on currently available information. For most licensees, this meant that the OIP used the current design-basis information for hazard evaluation. Coincident with the issuance of Order EA-12-049, on March 12, 2012, the NRC staff issued a Request for Information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.54(f) [Reference 20] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies. Due to the time needed to reevaluate the hazards, and for the NRC to review and approve them, the reevaluated hazards were generally not available until after the mitigation strategies had been developed. The NRC staff has developed a proposed rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was published for comment in the *Federal Register* on November 13, 2015 [Reference 47]. The proposed MBDBE rule would make the intent of Orders EA-12-049 and EA-12-051 generically applicable to all present and future power reactor licensees, while also requiring that licensees consider the reevaluated hazard information developed in response to the 50.54(f) letter.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in response to the 50.54(f) letter and the requirements for Order EA-12-049 and the MBDBE rulemaking (see COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards" [Reference 44]. The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 21]. The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 [Reference 34], the NRC staff informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the NRC SEs and inspections related to Order EA-12-049 will rely on the guidance provided in JLD-ISG-2012-01, Revision 0, and the related industry guidance in NEI 12-06, Revision 0. The hazard reevaluations may also identify issues to be entered into the licensee's corrective action program consistent with the OIPs submitted in accordance with Order EA-12-049.

As discussed above, licensees are reevaluating the site seismic and flood hazards as requested in the NRC's 50.54(f) letter. After the NRC staff approves the reevaluated hazards, licensees will use this information to perform flood and seismic mitigating strategies assessments (MSAs) per the guidance in NEI 12-06, Revision 2, Appendices G and H [Reference 48]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 49]. The licensee's MSAs will evaluate the mitigating strategies described in this SE using the revised seismic hazard information and, if necessary, make changes to the strategies or equipment. Licensees will submit the MSAs for NRC staff review.

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

3.5.1 Seismic

In its FIP, the licensee stated that seismic hazards are applicable to the site. The FIP also states that, per UFSAR Section 2.5, the safe shutdown earthquake (SSE) seismic criteria for the site is 15 percent of the acceleration due to gravity (0.15g) peak horizontal ground acceleration and 0.10g peak ground acceleration acting vertically. It should be noted that the actual seismic hazard involves a spectral graph of the acceleration versus the frequency of the motion. Peak acceleration in the frequency range that affects structures, such as the numbers above, is often used as a shortened way to describe the hazard.

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

3.5.2 Flooding

In its FIP, the licensee stated that CNS is subject to external flooding from Probable Maximum Floods (PMFs) resulting from Probable Maximum Precipitation events, Standard Project Floods (SPFs) equal to one half of the PMF, failures of upstream dams, a combination of dam failures and SPF's, seiche, hurricanes, and storm surge.

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

3.5.3 High Winds

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

The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06, Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009). If the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 mph exceeds $1x10^{-6}$/year, 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, 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 $1x10^{-6}$/year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 35° 03' 05" north latitude and 081° 04' 10" west longitude. In NEI 12-06, Figure 7-1, Contours of Peak-Gust Wind Speeds, Annual Exceedance Probability of $1x10^{-6}$/year, the location of CNS has a hurricane peak-gust wind speed of 150-160 mph. In NEI 12-06 Figure 7-2, Recommended Tornado Design Wind Speeds for the $1x10^{-6}$/year Probability Level indicates the site is in Region 1, and indicates a recommended tornado design wind speed of 172 mph for the site location. Therefore, the plant screens in for an assessment for high winds and tornados, including missiles produced by these events.

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

3.5.4 Snow, Ice, and Extreme Cold

As discussed in NEI 12-06, Section 8.2.1, all sites should consider the temperature ranges and weather conditions for their site in storing and deploying 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.

From UFSAR Table 2-29, Catawba Nuclear Station Vicinity Climatology, the lowest, average daily minimum temperature on a monthly basis is 32.1°F and occurs in January with the record lowest temperature of -5°F occurring in February. The maximum 24-hour snowfall was 16.5 inches, occurring in February. Section 2.3.1.2 in the UFSAR, Regional Meteorological Conditions for Design and Operating Bases, describes that winter conditions as a rule are not conducive to the development of major snow storms. Long-term records for the area show highest 24-hour snowfall near 18 inches (Winston-Salem, N.C., December, 1930). Ice storms, a much more frequent occurrence, do cause considerable damage over limited areas and can be expected several times a year. Typical accumulations range between one-quarter to one half inch.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 35° 03' 05" north latitude and 081° 04' 10" west longitude. In addition, the NRC staff notes that the site is located within the region characterized by Electric Power Research Institute (EPRI) as ice severity level 5 (NEI 12-06, Figure 8-2, Maximum Ice Storm Severity Maps). Consequently, the site is subject to extreme icing conditions that could also cause catastrophic destruction to electrical transmission lines. The licensee concludes that the plant screens in for an assessment for snow, ice, and extreme cold hazard.

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

3.5.5 Extreme Heat

In the section of its FIP regarding the determination of applicable extreme external hazards, the licensee stated that, as per NEI 12-06 Section 9.2, all sites are required to consider the impact of extreme high temperatures. From UFSAR Table 2-29, Catawba Nuclear Station Vicinity Climatology, the highest, average daily maximum temperature on a monthly basis is 88.3°F and occurs in July and the record highest temperature of 104°F occurred in September. Each month from May to September has record high temperatures of 100°F or higher. The plant site screens in for an assessment for extreme high temperature hazard.

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

3.5.6 Conclusions

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

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

In its FIP, the licensee described that CNS has implemented the single structure FLEX storage option utilizing one structure (the FSF) for storing FLEX equipment that is located on-site and protects FLEX equipment from all applicable hazards. The FSF contains sufficient equipment to satisfy redundancy requirements of NEI 12-06, Revision 0 for reliability and availability. The FSF is a dome structure that has an outside diameter of 144 feet. It has two equipment doors on opposite sides of the building and two personnel access doors. The building is located outside of the protected area approximately 500 feet west of the shipping and receiving warehouse, in a portion of the material lay-down storage area. FLEX equipment stored in the FSF includes low pressure pumps, medium pressure pumps, high pressure pumps, 120 Vac DGs, 600 Vac DGs, hose trailers, a fuel transfer trailer, a CAT 924K, a pickup truck, portable spot coolers, 600 Vac sump pumps, 120 Vac sump pumps, portable transformers, portable panelboards, ventilation fans, and 230 Vac sump pumps.

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

3.6.1.1 Seismic

In its FIP, the licensee stated that the FSF was seismically designed in accordance with the considerations presented in ASCE 7-10, and also meets the CNS SSE criteria. During the audit the NRC staff reviewed the licensee corporate FLEX program plan (PD-OP-ALL-0500, "Diverse and Flexible Strategies (FLEX) Program," Revision 1), which is applicable to CNS. This program plan requires that large portable equipment be secured as appropriate to withstand up to a SSE level seismic event and that stored equipment and structures be evaluated and protected from seismic interactions.

3.6.1.2 Flooding

In its FIP, the licensee described that the FSF at CNS is located above the flood level; thus it is protected from impacts due to flooding.

3.6.1.3 High Winds

In its FIP, the licensee described that the FSF is designed to resist wind forces and tornado missiles of a magnitude that bounds all design basis hazards.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee stated that because CNS is in a temperate climate, storage and operation of the FLEX equipment in high and low temperatures is not considered to be an issue. During the audit the NRC staff reviewed the licensee's corporate FLEX program, PD-OP-ALL-0500, Revision 1, and verified that provisions for equipment storage address the concerns of snow, ice extreme cold and extreme heat. Specifically, the program plan states that FLEX equipment storage will be such that the equipment will be maintained at temperatures (low and high) such that it is likely to function when called upon. In addition the program plan states that the FLEX storage facilities will be designed for the site design basis snow, ice and cold conditions.

3.6.2 Reliability of FLEX Equipment

Section 3.2.2 of NEI 12-06 states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare (i.e., an "N+1" capability, where "N" is the number of units on site). It is also acceptable to have a single resource that is sized to support the required functions for multiple units at a site (e.g., a single pump capable of all water supply functions for a dual unit site). In this case, the "N+1" could simply involve a second pump of equivalent capability. In addition, it is also acceptable to have multiple strategies to accomplish a function, in which case the equipment associated with each strategy does not require an additional spare.

As described in the FIP, major FLEX equipment includes the following:

- Low Pressure Pumps – the CNS FLEX strategy relies on a portable, diesel-driven, low pressure, high volume pump to supply water from the SNSWP to the FLEX raw water distribution system. Each FLEX low pressure pump is a portable, diesel-driven pump that can supply a design flow of 3,000 gpm which can meet the demands of both units simultaneously. The FLEX low pressure pump provides water to the FLEX medium pressure pumps via hoses (and also fire protection piping, if desired). CNS has two portable FLEX low pressure pumps to satisfy the "N+1" provision.

- Medium Pressure Pumps - the CNS FLEX strategy relies on a portable medium pressure pump to provide makeup water to the SGs. The FLEX medium pressure makeup pump can also be used to provide RCS makeup if the event occurs when the plant is in modes 5 or 6. The FLEX medium pressure pumps are portable, diesel-driven centrifugal pumps that can supply 300 gpm at a maximum pressure of 400 psig. CNS has three FLEX medium pressure pumps to satisfy the "N+1" provision.

- High Pressure Pumps - for an ELAP event initiating in Modes 1 - 4, the CNS FLEX strategy relies on a high pressure pump to provide RCS makeup and boration. The FLEX high pressure pumps are portable, diesel-driven, centrifugal pumps that can supply 40 gpm at 1700 psig, which is adequate to support the reactivity control and RCS system make-up requirements for the FLEX strategy. CNS has three portable FLEX high pressure pumps to satisfy the "N+1" provision.

- TDAFW Pit Portable Sump Pumps - a portable sump pump is placed in the TDAFW pit to pump out normal drains input from the TDAFW pump to the room sump before flooding impacts operation of the TDAFW pump. Each sump pump is electrically powered and requires 120 Vac, which can be supplied either via the FLEX electrical distribution system, or a 120 Vac DG. The sump pumps can deliver 15 gpm flow at over 30 feet of head, which is adequate to prevent flooding of the TDAFW pit sumps.

CNS has three 120 Vac sump pumps, which are sufficient to support the CNS FLEX strategies.

- Auxiliary Building Portable Sump Pumps - CNS can deploy portable 600 Vac sump pumps to manage internal flooding from potential pipe breaks. Each 600 Vac sump pump can be powered by one of the 600 Vac DGs, although a single DG may not have sufficient capacity for operation of all three sump pumps simultaneously. CNS has three 600 Vac sump pumps, which are sufficient to support the CNS FLEX strategies.

- Diesel Generators - CNS will deploy portable 600 Vac FLEX DGs (one for each unit), and associated support equipment to establish the FLEX electrical distribution system. CNS has three FLEX 600 Vac DGs to satisfy the "N+1" provision.

By letter dated September 27, 2016, the licensee indicated that they intend to follow alternative guidance for hoses and cables, as further discussed in Section 3.14.1 of this SE. Other than this alternative, based on the number of portable FLEX pumps, FLEX DGs, and support equipment identified in the FIP, 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 RCS makeup and boration, SFP makeup, and maintaining containment consistent with the "N+1" provision in Section 3.2.2 of NEI 12-06.

3.6.3 Conclusions

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

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee stated that CNS will use pre-defined deployment routes to transport FLEX equipment to the staging areas. The identified paths and deployment equipment positions will be accessible during all modes of operation. Personnel will periodically perform walk downs of the deployment paths to ensure pathways remain clear.

3.7.1 Means of Deployment

In its FIP, the licensee described that two deployment teams will be dispatched to transport Phase 2 FLEX equipment from the FSF to deployment locations. One team will use the CAT 924 loader and the other will use a Dodge 5500 truck. Catawba has developed guidance on an appropriate sequence of deployment actions for each team. The deployment routes for FLEX equipment begin with passage through the vehicle barrier access port and proceed along five primary on-site deployment paths described in detail in the FIP. The licensee's CAT 924 front loader is intended for debris clearing on the deployment paths and towing heavy equipment.

3.7.2 Deployment Strategies

In its FIP, the licensee stated that the deployment paths were evaluated for seismic stability and liquefaction and determined to be acceptable. A flooding event may inundate portions of the site, but flood waters will recede from all deployment paths and staging areas in time for FLEX deployment to meet all time constraints.

Pre-determined deployment and staging locations for FLEX equipment necessary to support the FLEX strategies have been identified and are summarized as follows:

Makeup water for the auxiliary feedwater system to the SGs via the TDAFW pump will be supplied by the FLEX low pressure pump, which will be staged on a ramp at the edge of the SNSWP. Suction hoses with strainers will be attached to the pump and placed into the SNSWP. Based on the CNS UFSAR, Section 2.4, using historical information, the licensee does not expect water temperature in the SNSWP to decrease to the point where access will be challenged due to ice formation.

SG feedwater may be delivered by a FLEX medium pressure pump that draws suction from the discharge of the low pressure pump. The Unit 1 medium pressure pump is normally deployed south of the Unit 1 exterior SG Doghouse door. The Unit 2 medium pressure pump is normally deployed north of the Unit 2 FWST and west of the Unit 2 containment mechanical equipment building. Discharge hoses from the FLEX medium pressure pump will be routed to the auxiliary feedwater system piping connections located in the exterior and interior SG Doghouses. For Phase 3 deployment, procedures and FSGs were developed to connect NSRC equipment to station equipment. Use of portable pumps to supply feedwater to the SGs remains the same as Phase 2 until the RHR system is restored to service with supplemental power from the NSRC DGs or off-site power is restored.

Two FLEX high pressure pumps (one for each unit) with hoses/adapters will be deployed to support RCS boration and inventory control. For Unit 1, the normal staging location for the FLEX high pressure pump is south and west of the access to Unit 1 electrical penetration room. For Unit 2, the normal staging location for the FLEX high pressure pump is north of the access to Unit 2 electrical penetration room. Suction hoses will be deployed to connect the portable pump to the SFP cooling system connections from the FWST. A single FWST will supply borated water to the portable pumps for both units using a gated wye connection. Discharge hoses from the FLEX high pressure pumps will be connected to the discharge piping of the safety injection pumps in the Auxiliary Building to supply injection makeup to the RCS.

Two 600 Vac portable DGs and cabling will be deployed to an area near the FWSTs of each unit (one generator per Unit). The normal staging area for the Unit 1 FLEX 600 Vac DG is east of the Unit 1 turbine building, in close proximity to the Unit 1 motor generator (MG) set room door. The normal staging area for the Unit 2 FLEX 600 Vac DG is east of the Unit 2 turbine building, in close proximity to the Unit 2 MG set room door. Alternate staging areas are near the waste shipping/auxiliary access point area doors, the hot machine shop area doors, and the Unit 1 condenser circulating water system pit/safe shutdown facility area. The portable 600 Vac DGs will be connected to the normal plant MCCs for re-powering equipment via the FLEX backbone using a combination of permanently installed and portable cables, portable panelboards, and transformers. Cables and connectors of the FLEX 600 Vac distribution system are color coded to ensure proper phase rotation.

During Phase 3, DGs from the NSRC will be available to provide additional power. For Unit 1, the NSRC 4160 Vac DGs and switchgear will be staged south of the diesel generator building and the NSRC 480 Vac DGs will be staged south and west of the Auxiliary Building, near the FLEX high pressure pump. For Unit 2, the NSRC 4160 Vac DGs and switchgear and the NSRC 480 Vac DGs will be staged near the west side of the diesel generator building.

Catawba can deploy FLEX sump pumps to ensure that equipment for core cooling is not compromised by flooding. The normal FLEX sump pump staging area is at the rear of the Auxiliary Building near the clean trash room. Alternate staging areas include the area outside the Unit 1 MG set room and the area outside the Unit 2 MG set room.

Eleven portable 6 kW DGs are available to be deployed from the FSF to the location where they are needed. These DGs will be used to power battery chargers for hand-held radios and portable satellite phones, fans, small sump pumps, and other identified loads.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling (SG) Primary and Alternate Connections

In its FIP, the licensee described the primary connections for SG makeup as being located on auxiliary feedwater system piping downstream of the containment isolation valves. The licensee stated that the four connection points in the auxiliary feedwater system piping correspond to one connection for each of the four SGs. At each of the four connection points, the licensee will replace a blind flange with a hose adapter to establish a flow path. The four auxiliary feedwater system piping connections are located inside the SG Doghouses, which are Seismic Category I structures and, according to the licensee's FIP, are protected from the applicable external hazards. The licensee also described the alternate SG makeup connections as being located on Steam Generator Wet Layup Recirculation system piping, and are also located in the SG Doghouses. As with the connections on the auxiliary feedwater system piping, the four connection points for the steam generator wet layup recirculation system provide one connection for each of the four SGs. The licensee will also replace the blind flange with a hose adapter to establish a flow path at each of the four alternate connection points. According to the licensee's FIP these connection points are protected from the applicable external hazards. By letter dated February 28, 2014 [Reference 12], the licensee clarified that the alternate connections have been evaluated to be seismically robust but could be unavailable for certain flooding scenarios. NEI 12-06, Section 3.2.2 stipulates that both the primary and alternate connection points do not need to be available for all applicable hazards, but the location of the connection points should provide reasonable assurance of at least one connection being available. Since the primary connection points are available for all external hazards applicable to CNS, and the alternate connection points are available for all but a possible flooding scenario, the NRC staff finds that the licensee has met the NEI 12-06 provisions for these connections.

The licensee also described in the FIP the capability to supply the TDAFW pump directly from the nuclear service water system during Phase 3. The primary connection utilizes the train "B" of the nuclear service water system supply header through a fill valve. The fill valve is located in the Nuclear Service Water Pumphouse, which is a Seismic Category I structure and is protected from all external hazards. The alternate connection to the nuclear service water system is found on train "A". The connection is through an access plug, which is located inside the protected area fence near the Nuclear Service Water Pumphouse.

RCS Inventory Control Primary and Alternate Connections

In its FIP, the licensee described the primary connection for RCS makeup as the connection on the FWST supply line providing the suction supply for the portable high pressure pump. The FWST supply line normally is used to supply water to the SFP, however the licensee will use this portion to connect the portable high pressure pump and divert the FWST water to the RCS. The licensee stated that the connection point is in the Auxiliary Building, which is a Seismic Category I structure and protected from all external hazards. The licensee also stated that the FWST of each unit will serve as the primary suction source for both units. The other unit FWST will serve as the backup source. The portable high pressure pump will be connected to a gated wye assembly, with the flexibility to utilize either unit as necessary. The licensee also described in its FIP, additional RCS makeup connections. The licensee stated that connection points on the safety-related discharge piping of the safety injection system will supply borated water to the RCS using the high pressure makeup pump. The licensee also stated that the connections are available on Train "A" and Train "B", which serve as primary and alternate connection points respectively. The safety injection piping is located in a Seismic Category I structure and is protected from all external hazards.

SFP Makeup Primary and Alternate Connections

The licensee described in its FIP, the primary strategy SFP makeup, which is the pressurizing of Train "B" of the nuclear service water system using the FLEX low pressure pump. Train "B" of the nuclear service water system is located inside the Nuclear Service Water Pumphouse. The Nuclear Service Water Pumphouse, as described above, is a Seismic Category I structure and is protected from all external hazards. The licensee indicated that Train "A" of the nuclear service water system will provide an alternate connection for SFP makeup. The associated valves for the nuclear service water system and SFP cooling system are to be manually operated inside the Auxiliary Building, which is a Seismic Category I structure and protected from all external hazards. The licensee also described an additional alternate strategy for SFP makeup, which uses a jumper hose to connect the nuclear service water system to the SFP cooling system skimmer loop. The SFP cooling system skimmer loop flows through a manifold line around the SFP with a series of discharge points controlled by manual valves.

3.7.3.2 Electrical Connection Points

The CNS FLEX strategy for re-powering the 600 Vac vital bus circuits during Phase 2 is through the use of a FLEX Backbone (pre-installed distribution panels, panelboards, cables) and the deployment of one of the 600 Vac FLEX DG per unit connected to the backbone distribution panels. The 600 Vac FLEX DG allows for recharging Class 1E batteries, maintaining vital instrumentation, and repowering plant equipment. Operators would deploy the portable 600 Vac DGs, panelboards, and associated cabling from the FSF to one of five identified locations. The primary locations for staging is east of both units turbine building in close proximity to their respective MG set room doors. The alternate locations are near the waste shipping/auxiliary access point area doors, the hot machine shop area doors, and the CNS, Unit I condenser circulating water pit/SSF area.

Once the FLEX DGs are positioned in the selected locations, the licensee will set up panelboards and connect them to the distribution panels. The panelboards will be connected to normal plant MCCs for repowering equipment via the FLEX backbone. This strategy utilizes MCCs 1EMXG, 2EMXH, 1(2)EMXA, 1(2)EMXI, 1(2)EMXB, 1(2)EMXJ, 1(2)EMXK, 1(2)EMXS, 1(2)EMXC, 1(2)EMXL, and 1(2)EMXD. The alternate connection strategy will require the use of a spare portable panelboard that will directly feed four selected MCCs from the FLEX DGs without going through the distribution panel. The licensees FLEX guideline, FSG-20, "FLEX Electrical Distribution," Revision 3, provides direction for repowering MCCs and ensuring proper phase rotation before attempting to power equipment from the 600 Vac FLEX DG.

For Phase 3, the licensee will receive two (per unit) 1-MW 4160 Vac and one (per unit) 1100 kW 480 Vac combustion turbine generators from the NSRC. The two 1-MW 4160 Vac combustion turbine generators would be connected to the appropriate 4160 VAC busses in order to meet the required 4160 Vac load requirements for each unit. For Unit 1, the NSRC 4160 Vac turbine generators will be staged south of the Diesel Generator Building and the NSRC 480 Vac turbine generator will be staged south and west of the Auxiliary Building, near the FLEX high pressure pump. For Unit 2, the NSRC 4160 Vac and 480 Vac turbine generators will be staged near the west side of the Diesel Generator Building. The 480 Vac to 600 Vac step-up transformer will also be deployed in the vicinity of the 480 Vac turbine generators. The licensee's FLEX guidelines for both units, generically titled FSG-23, "Long Term FLEX Strategies," provide direction for ensuring proper phase rotation before attempting to power equipment from the 4160 Vac and 480 Vac combustion turbine generators.

3.7.4 Accessibility and Lighting

With regard to lighting, in its FIP the licensee stated that post-fire safe shutdown lighting is available in many areas where manual actions are necessary. The post-fire safe shutdown lights have self-contained batteries with an 8-hour life. Additional portable lighting will be provided for use in the yard.

Lighting units included in the FLEX strategy are as follows: light-emitting diode (LED) tripod mounted lights, LED string lights, and miscellaneous helmet lights and flashlights.

The LED-mounted tripods will be deployed in many areas including, the MCR, auxiliary feedwater system pump room(s), MG set room(s), interior and exterior SG Doghouses, electrical penetration rooms, battery room(s), Technical Support Center (TSC), and general floor areas in the Auxiliary Building. String lights will also be used in the MCR and the TSC. The lighting plan includes a total of 27 LED-mounted tripods and 3 string lights.

3.7.5 Access to Protected and Vital Areas

According to the licensees FIP, access through security fences, doors, and other barriers will be unimpeded. During the audit process, the licensee provided information describing that contingencies are in place to ensure that access to protected and vital areas will not be hindered if the normal access control systems are without power.

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee described that diesel fuel oil for the FLEX equipment will be obtained from the safety-related, underground emergency diesel generator (EDG) fuel oil storage tanks. The fuel will be pumped out of the underground tanks and transferred to the portable diesel fuel storage tank using a diesel-driven portable transfer pump or will be gravity drained into small containers in the EDG room and hand carried to the location needed. Connections on the piping from the EDG tanks that will be used for removing fuel oil are seismically qualified and located above the flood level in the yard. The portable fuel tank will be pulled by the FLEX pickup truck or other available vehicles to the various site locations for refueling portable diesel generators.

The estimated fuel volume required to support FLEX equipment for 24 hours is 2,027 gallons. FLEX equipment will be stored full of diesel fuel oil and refueled as necessary. The CNS fuel consumption calculation determined that 485 gallons of diesel fuel oil would be required for refueling in the first 24 hours, which is well within the capacity of the EDG diesel fuel oil storage tanks.

To preclude multiple pieces of FLEX equipment running out of fuel at the same time, personnel will attempt to maintain individual fuel tanks greater than half full. The small DGs may require refueling prior to the FLEX diesel fuel transfer trailer being available. If necessary, personnel will transport required fuel by hand.

During Phase 3, additional diesel fuel will be brought onsite from outside resources as required.

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

3.8 Considerations in Using Offsite Resources

3.8.1 CNS SAFER Plan

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

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

By letter dated September 26, 2014 [Reference 22], the NRC staff issued its staff assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the staff concluded that SAFER has procured equipment, implemented appropriate processes to maintain the equipment, and developed plans to deliver the equipment needed to support site responses to BDBEEs, consistent with NEI 12-06 guidance; therefore, the staff concluded in its assessment that licensees can reference the SAFER program and implement their SAFER Response Plans to meet the Phase 3 requirements of Order EA-12-049.

In its FIP, the licensee stated that the SAFER Response Plan for CNS contains: (1) SAFER control center procedures, (2) National SAFER Response Center procedures, (3) logistics and transportation procedures, (4) staging area procedures, which include travel routes between staging areas to the site, (5) guidance for site interface procedure development, and (6) a listing of site-specific equipment (generic and non-generic) to be deployed for FLEX Phase 3. During the audit review the NRC staff confirmed that CNS had developed a SAFER response plan that included the applicable site-specific information.

3.8.2 Staging Areas

In general, up to four staging areas for NSRC supplied Phase 3 equipment are identified in the SAFER Plans for each reactor site. These are a primary (Area "C") and an alternate (Area "D", if needed) which are offsite areas (within about 25 miles of the plant) for receipt of ground transported or airlifted equipment from the SAFER centers in Phoenix, Arizona or Memphis, Tennessee. From Staging Areas "C" and/or "D", a near-site or on-site staging area "B" is established for interim staging of equipment prior to it being transported to the final location for implementation in Phase 3 at Staging Area "A".

For CNS, alternate staging area "D" is not used. Staging Area "C" is the Kings Mountain Training Center (24 air miles and 34 road miles from CNS), Staging Area "B" is an on-site parking lot outside the protected area. Staging Area "A" consists of several locations for final equipment set up as described in Section 3.7.2, above.

Primary and alternate driving routes from staging area "C" to Staging Area "B" have been identified. Catawba will coordinate with local and state authorities to assess the condition of roads and bridges along the travel path. If ground transportation from Staging Area "C" to Staging Area "B" is not feasible, NSRC equipment can be delivered to Staging Area "B" by helicopter airlift. Two access routes from Concord Road to Staging Area "B" have been identified: the primary access location is through the main entrance, northwest gate; the secondary access path is through the main entrance, southwest gate.

3.8.3 Conclusions

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

3.9 Habitability and Operations

3.9.1 Equipment Operating Conditions

3.9.1.1 Loss of Ventilation and Cooling

Following a BDBEE and subsequent ELAP event at CNS ventilation providing cooling to occupied areas and areas containing FLEX strategy equipment will be lost. Per the guidance given in NEI 12-06, FLEX strategies must be capable of execution under the adverse conditions (unavailability of installed plant lighting, ventilation, etc.) expected following a BDBEE resulting in an ELAP/loss of normal access to the UHS. The primary concern with regard to ventilation is the heat buildup which occurs with the loss of forced ventilation in areas that continue to have heat loads. The key areas identified for all phases of execution of the FLEX strategy activities are the MCR, TDAFW pump room, SFP area, vital battery rooms, electrical switchgear rooms, cable area (Unit 1/2 and Electrical Panelboard Cubicles), and containment. The licensee evaluated these areas to determine the maximum steady state temperatures expected in specific areas related to FLEX implementation to ensure the environmental conditions remain acceptable for personnel habitability and equipment operation.

The NRC staff reviewed calculations NAI-1813-001, "Catawba Auxiliary Building Extended Loss of AC Power FLEX Response," Revision 0, NAI-1813-002, "Catawba Fuel Handling Building Extended Loss of AC Power FLEX Response," Revision 0, and CNC-1211.00-00-0036, "SSF Temperature Calculation/Auxiliary Feedwater Pump Room Area," Revision 1, to verify equipment will not be adversely affected by increases in area temperatures as a result of the loss of ventilation. Based on the temperatures calculated, the NRC staff finds that personnel performing actions in those areas would not be adversely affected by local area temperatures caused by the loss of ventilation. The NRC staff also reviewed FSG-05, "Initial Assessment and FLEX Equipment Staging," Revision 4 which provides guidance for opening doors and deploying portable fans after a loss of ventilation and cooling and FSG-11, "Alternate Spent Fuel Pool Makeup and Cooling," Revision 1, to confirm consistency with the analyses assumptions.

MCR

For the MCR, the licensee's calculation (NAI-1813-001) showed that the temperature will remain below 90°F if four spot coolers are deployed and powered by the portable FLEX diesel generators within 8 hours. Administrative controls for opening of the Control Room doors is contained in ECA-0.0, “Loss of All AC Power”, Revision 49 (Revision 48, Unit 2). Additionally, FSG-5, “Initial Assessment and FLEX Equipment Staging,” Revision 4, discusses deployment of the spot coolers and HVAC units.

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

TDAFW Pump Room

The licensee’s FIP states that analysis validated TDAFW pump room temperatures remain acceptable with no additional [operator] action under ELAP/loss of access to the UHS conditions. During the audit process the NRC staff reviewed calculation NAI-1813-001, which showed that the TDAFW pit area does not exceed 121.3°F for the 48 hours the TDAFW pump is assumed to operate. This is less than the allowable room temperature of 145°F.

Based on temperatures remaining below the established limit for TDAFW components during the pump’s mission time, the NRC staff finds that the equipment in the TDAFW pump room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Vital Battery Room

The CNS safety-related batteries are manufactured by GNB. Calculation (NAI-1813-001) shows that the temperature in the vital battery rooms remains less than 120°F. The battery vendor’s analysis shows that the batteries are capable of performing their function up to 120°F, however, periodic monitoring of electrolyte level would be necessary to protect the battery since the battery may gas more at higher temperatures. The concrete walls, floors and ceiling in the safety-related battery rooms function as a large heat sink following loss of ventilation. The licensee plans to open all doors in the vital battery area, which include the doors to each battery cubicle and the doors in the wall separating the Unit 1 Vital Battery Area from the Unit 2 Vital Battery Area. An outdoor flow path is created by pulling air through the MG set room, through the Unit 2 Auxiliary Building stairwell, pushed through the Unit 1 and 2 Vital Battery Areas, through the Service Building, up through the Service Building Stairwell shaft, and to the outdoors through an open door at the top of the Service Building Stairwell shaft.

The Vital Battery Room analysis contained in calculation NAI-1813-001 evaluated two heat-up scenarios: unmitigated and mitigated. The unmitigated case showed that the maximum temperature reached in the Vital Battery Room in the first 120 hours of an ELAP event was approximately 120°F. By taking the mitigating action of opening doors 24 hours after the ELAP-initiating event, the maximum temperature was shown to be reduced to 118.1°F. Guideline FSG-05 provides direction for opening battery room doors and establishing portable ventilation in the vital battery rooms.

Based on the above, the NRC staff finds that the licensee’s ventilation strategy, will maintain the battery room temperature below the maximum temperature limit (120°F) of the batteries.

Therefore, the NRC staff finds that the CNS safety-related batteries should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP event.

Electrical Switchgear Rooms and Cable Area (Unit 1/2 and Electrical Panelboard Cubicles)

For the electrical switchgear rooms and cable area, calculation (NAI-1813-001) showed that the temperature will remain below 120°F with no mitigating actions taken with the exception of the Unit 1 Electrical Panel Board Room. The licensee plans to open the Electrical Panel Board Room doors within the first 24 hours of the event to maintain temperature less than 120°F.

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

Containment

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

3.9.1.2 Loss of Heating

In its FIP, the licensee stated they will take appropriate actions to ensure protection of the FLEX strategies from cold conditions. This includes starting FLEX diesel-powered equipment early due to the potential for difficult starting in extreme cold conditions and establishing trickle drains from idle equipment and hoses to prevent formation of ice plugs. The license also stated that they will provide heat tracing for equipment in the FSF, FWST level instrumentation, and pressure instruments in the SG Doghouses.

The battery rooms are in the interior of the Auxiliary Building and are normally maintained at approximately 75 to 80°F. In the event of an ELAP, the battery room temperatures are expected to rise with loss of ventilation. Therefore, reaching the minimum pilot cell temperature limit of 65°F is not expected. Based on its review of the licensee's battery room assessment, the NRC staff finds that the safety-related batteries should perform their required functions at the expected temperatures as a result of loss of heating during an ELAP event.

The licensee indicated in its FIP that heat tracing would not be needed to implement FLEX strategies on site. During the audit, the licensee provided evaluations showing that the FLEX connections related to the FWST recirculation piping are insulated and the recirculating of the FWST water provides freeze protection during cold weather under normal conditions. FLEX strategies being implemented within 8-12 hours as indicated in the FLEX timeline would provide FWST circulation needed to prevent the freezing of the FWST recirculation piping. The licensee also stated as part of its evaluation that no critical instrumentation in the SG Doghouses would be subject to freezing. The SG Doghouses are protected due to insulation and the heat of the steam piping located there. The instrumentation in the Auxiliary Building and Annulus will not be subject to freezing due to thermal inertia and the loss of forced ventilation. The SFP will also remain above freezing due to the decay heat from the spent fuel in the SFP. During the audit, the NRC staff observed that piping and instrumentation necessary for implementation of the licensee's plan is generally located in areas that should not require heat tracing for cold weather conditions.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3, is the potential buildup of hydrogen in the battery rooms as a result of loss of ventilation during an ELAP event. The NRC staff reviewed CNS calculation, NAI-1813-001, "Catawba Auxiliary Building Extended Loss of AC Power FLEX Response," Revision 0, to verify that hydrogen gas accumulation in the 125 Vdc Vital Battery rooms will not reach combustible levels while HVAC is lost during an ELAP. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. In order to prevent a buildup of hydrogen in the battery rooms, the licensee's procedure FSG-05, "Initial Assessment and FLEX Equipment Staging," direct plant operators to open the battery room doors and establish ventilation using portable fans to disperse the off-gassing hydrogen to a much larger area in order to prevent any significant hydrogen accumulation. The licensee stated in its FIP, that the battery rooms will be ventilated by pulling outside air through the Unit 2 MG set room, down the Unit 2 Auxiliary Building stairwell, through the battery rooms, and up the Service Building stairwell to outdoors through an open door creating a chimney effect.

Based on its review of the licensee's calculation and battery room ventilation strategy, the NRC staff finds that hydrogen accumulation in the CNS safety-related battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP as a result of a BDBEE since the licensee plans to open battery room doors and place portable fans in service when the battery chargers are repowered during Phase 2 and Phase 3.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

As stated in the FIP and demonstrated in calculation NAI-1813-001, selected doors will be opened and spot coolers deployed to reduce and maintain temperatures at about 90 °F. Administrative controls for opening of the Control Room doors is contained in ECA-0.0, "Loss of All AC Power", Revision 49 (Revision 48, Unit 2). Additionally, FSG-5, "Initial Assessment and FLEX Equipment Staging", Revision 4, discusses deployment of the spot coolers and HVAC units. Opening of doors and the deployment of spot coolers is not based on any time limit but on limiting room temperature to 90°F. Thus, habitability conditions should be sufficient to support continuous occupancy for the operators to perform the strategies. The licensee's analysis and plans of opening the MCR doors, providing spot coolers and FLEX HVAC units should maintain the Control Room temperature below the conservative limit for MCR habitability of 110°F, which was analyzed in NUMARC 87-00.

3.9.2.2 Spent Fuel Pool Area

As discussed in Section 3.3.4.1.1 of this SE, a ventilation path will be established by opening selected doors to allow steam to escape from the SFP area within 6 hours following an ELAP-initiating event. FSG-05 and FSG-11 provide guidance on opening doors for ventilation. The conditions for different locations on the refueling floor are calculated in NAI-1813-002, "Catawba Fuel Handling Building Loss of AC Power FLEX Response," Revision 0. The FIP states that the nominal minimum time to boil for a non-outage situation is approximately 37 hours after the initiating event. Thus, the actions required to employ the SFP cooling strategy are anticipated to take place long before the conditions become challenging due to habitability considerations.

3.9.2.3 Other Plant Areas

The FIP did not identify any other rooms or areas which would require continuous occupancy. The NRC staff considered areas that may require temporary occupancy, such as the TDAFW pump rooms, the electrical penetration rooms, and the switchgear rooms. The staff reviewed the licensee's calculations NAI-1813-001, Revision 0, "Catawba Auxiliary Building Extended Loss of AC Power FLEX Response", and NAI-1813-002, Revision 0, "Catawba Fuel Handling Building Extended Loss of AC Power FLEX Response" and concluded that areas that may need temporary occupancy had acceptable temperature profiles for short-term entries.

3.9.3 Conclusions

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

3.10 Water Sources

3.10.1 Steam Generator Make-Up

In its FIP, the licensee stated that SG feedwater will be provided from any of the following sources:

- USTs
- CACST
- Hotwell
- condenser circulating water system piping embedded volume
- SNSWP

The embedded condenser circulating water system captured volume and the SNSWP are the credited sources of water because of their robustness to the applicable hazards. The USTs and the hotwell are not protected from external hazards. These tanks are normally aligned as a TDAFW pump suction source, but automatic realignment to embedded condenser circulating water system captured volume is provided if the default sources are not available.

3.10.2 Reactor Coolant System Make-Up

In its FIP, the licensee stated that for RCS boration during Phase 2, CNS will provide borated water from one or both of the following sources:

- FWSTs
- CLAs

The FWST is the credited source of makeup for the RCS. The minimum inventory of the intact FWST is 377,537 gallons and contains boron in accordance with CNS technical specifications. The FWST is seismically-qualified and the bottom portion is protected by a missile wall. If the top portion of the FWST is not damaged by the BDBEE, the initial inventory will be sufficient for RCS makeup for the duration of the event. If the top portion of the FWST is damaged, makeup to the FWST may be required within 52 hours. The CLAs are expected to inject a portion of their liquid volume during the plant cooldown and depressurization. Based on their safety-related classification, they are considered robust for the purposes of Order EA-12-049.

Alternate sources of borated water include the following options, none of which are robust:

- NSRC-supplied mobile boration skid
- Trucking from an off-site source (e.g., McGuire Nuclear Station)
- Recovery of borated water from the FWST annulus (if damaged)
- Portable FLEX drop tanks mixing boron and raw water

3.10.3 Spent Fuel Pool Make-Up

In its FIP, the licensee stated that for inventory control of the SFP, CNS uses raw water from the robust nuclear service water system, which is pressurized by the FLEX low pressure pump using the SNSWP as the suction source. The SNSWP will be available following the applicable extreme external hazards. During Phase 3, CNS may transition to a clean water source (e.g., NSRC-supplied water purification unit) when available.

3.10.4 Containment Cooling

In its FIP, the licensee described that the ice condenser containment design helps maintain containment conditions in all phases of the ELAP event, until the ice bed inventory is depleted. CNS relies on repowering a set of fans (hydrogen skimmer fans, lower containment ventilation system fans, and containment air return fans) to maintain containment temperature and pressure below acceptable limits. Containment spray functionality is not required to support CNS FLEX response strategies.

3.10.5 Conclusions

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

3.11 Shutdown and Refueling Analyses

Order EA-12-049 requires that licensees must be capable of implementing the mitigation strategies in all modes. In general, the discussion above focuses on an ELAP occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. When the ELAP occurs with the plant at power, the mitigation strategy initially focuses on the use of the steam-driven 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 reactor and placed in the SFP, there may be a shorter timeline to implement the makeup of water to the SFP. However, this is balanced by the fact that if immediate cooling is not required for the fuel in the reactor vessel, the operators can concentrate on providing makeup to the SFP. The licensee's analysis shows that following a full core offload to the SFP, about 48 hours are available to implement makeup before boil-off results in the water level in the SFP dropping to a level of 10 feet above the top of the fuel assemblies, and the licensee plans 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 steam-powered pump and allow operators to release steam from the SGs (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 35], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 36], the NRC staff endorsed this position paper as a means of meeting the requirements of the order.

The position paper provides guidance to licensees for reducing shutdown risk by incorporating FLEX equipment in the shutdown risk process and procedures. Considerations in the shutdown risk assessment process include maintaining necessary FLEX equipment readily available and potentially pre-deploying or pre-staging equipment to support maintaining or restoring key safety functions in the event of a loss of shutdown cooling. The NRC 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. By letter dated August 28, 2014 [Reference 13], the licensee informed the NRC staff of its plans to follow the guidance in this position paper. During the audit process, the NRC staff observed that the licensee had made progress in implementing this position paper.

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

3.12 Procedures and Training

The licensee has developed a comprehensive set of FSGs. In its FIP, the licensee stated 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 response strategies for accomplishing specific tasks in support of EOPs and Abnormal Operating Procedures. The FSGs are used to supplement (not replace) 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" to the extent necessary to include appropriate reference to FSGs and provide command and control for the ELAP.

In its FIP, the licensee stated that 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 was similarly weighted.

The NRC staff finds that the licensee has adequately addressed the procedures and training associated with FLEX. 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 to the NRC dated October 3, 2013 [Reference 37], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 38], the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs. In its FIP, the licensee stated that they would conduct maintenance and testing of the FLEX equipment in accordance with the industry letter.

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

3.14.1 Reduced Set of Hoses and Cables

By letter dated September 27, 2016 [Reference 58], the licensee indicated their plan to implement an alternative approach to the NEI 12-06, Revision 0, guidance for hoses and cables. In NEI 12-06, Section 3.2.2 states that in order to assure reliability and availability of the FLEX equipment required to meet these capabilities, 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. Thus, a single-unit site would nominally have at least two portable pumps, two sets of portable ac/dc power supplies, two sets of hoses & cables, etc. NEI, on behalf of the industry, submitted a letter to the NRC on May 1, 2015 [Reference 45], proposing an alternative regarding the quantity of spare hoses and cables to be stored on site. The alternative proposed was that that either a) 10 percent additional lengths of each type and size of hoses and cabling necessary for the "N" capability plus at least one spare of the longest single section/length of hose and cable be provided or b) that spare cabling and hose of sufficient length and sizing to replace the single longest run needed to support any FLEX strategy. By letter dated September 27, 2016 [Reference 58], the licensee stated that they had adopted the NEI proposal, as endorsed by the NRC. By letter dated May 18, 2015 [Reference 46], the NRC agreed that the alternative approach is reasonable, but that the licensees may need to provide additional justification regarding the acceptability of various cable and hose lengths with respect to voltage drops, and fluid flow resistance. Based on the licensee's use of the NEI guidance, as endorsed by the NRC, the staff approves this alternative for CNS as an acceptable method of compliance with Order EA-12-049.

3.14.2 Remove Requirement for SFP Spray

By letter dated September 27, 2016 [Reference 58], the licensee requested an alternative from NEI 12-06, Revision 0, Table D-3, which states that there should be three methods of adding water to the SFP to ensure that the spent fuel assemblies located there have sufficient cooling. The three methods are: (1) makeup using a hose directly into the SFP; (2) makeup using a connection to a SFP cooling piping or a similar system (which allows makeup while avoiding the need to go on the refueling floor, where habitability may be a concern); and (3) makeup via water spray into the SFP. The licensee has the first two capabilities, but asked for an alternative to not have the spray capability. The NRC's position as stated in JLD-ISG-2012-01, Revision 1 [Reference 49], is that the spray is needed if there is a potential for water to drain from the SFP.

By letter dated March 31, 2014 [Reference 50], the licensee submitted its Seismic Hazard Screening Report (SHSR) for CNS to the NRC in response to the NRC's 50.54(f) letter, [Reference 20] regarding re-evaluation of the seismic hazard at the site. The 50.54(f) letter requested that, in part, a seismic evaluation be made of the CNS SFP. More specifically, plants were asked to consider "...all seismically induced failures that can lead to draining of the SFP." The licensee's SHSR determined that since the ground motion response spectrum (GMRS) for CNS exceeds the SSE in the 1 to 10 Hertz (Hz) frequency range, that a SFP evaluation was merited. The NRC staff confirmed, by letter dated April 27, 2015 [Reference 51], that the GMRS exceeds the SSE in the frequency range of interest, and that a SFP evaluation is merited for CNS.

As part of the process to develop and evaluate responses to the 50.54(f) letter, the NRC staff worked with industry to develop a method of predicting the susceptibility of a SFP to cracking. The primary input is the seismic stresses that the SFP was designed to survive compared to the seismic stresses from a seismic event that has some probability of occurring at the site. By letter dated February 23, 2016 [Reference 52], NEI submitted a request for the NRC staff to approve guidance for a SFP evaluation based on the February 2016 EPRI report 3002007148, "Seismic Evaluation Guidance Spent Fuel Pool Integrity Evaluation." The EPRI report is a generic study performed for nuclear power plants with low-to-moderate seismic ground motions (peak spectral accelerations less than 0.8g) to address the NRC 50.54(f) letter that requested a seismic evaluation of the SFP to consider all seismically induced failures that could lead to rapid draining. Based on the NRC staff's assessment letter dated October 27, 2015 [Reference 53], the peak spectral acceleration for the licensee's SSE and GMRS are less than 0.8g; thus, the NRC staff finds that EPRI 3002007148 is applicable to the licensee's site. Section 3.3 of EPRI 3002007148 provides guidance on specific site parameters, structural parameters and non-structural parameters that a licensee should confirm on a site-specific basis to ensure that the report applies. The NRC staff endorsed the EPRI report as an acceptable method of performing seismic evaluations of SFPs in a letter dated March 17, 2016 [Reference 54].

By letter dated July 20, 2016 [Reference 55], the licensee confirmed that CNS met the parameters in Section 3.3 of the EPRI report to affirm that the report was applicable to the CNS SFP and therefore CNS was not susceptible to draining of the SFP. By letter dated August 11, 2016 [Reference 57], the NRC staff concluded that the licensee responded appropriately to the 50.54(f) letter regarding a SFP evaluation. Specifically, the NRC review noted that the licensee's submittal: (1) appropriately evaluated the CNS, Units 1 and 2 SFP structural capability, and (2) acceptably evaluated the non-structural considerations of the CNS, Units 1 and 2 SFPs, whose failure could lead to potential drain-down due to a seismic event.

The NRC staff reviewed the licensee's proposed alternative and finds that the robust design of the SFPs, the moderate seismic hazard at the site, and the available diverse methods of providing makeup to compensate for boil-off justify deleting the requirement for SFP spray as an acceptable alternative to the NEI 12-06, revision 0 guidance, consistent with the staff position presented in JLD-ISG-2012-01, Revision 1. The NRC staff concludes that the licensee has a strategy to maintain or restore SFP cooling that will prevent damage to the fuel following a BDBEE, which meets the requirement of the EA-12-049 order.

In conclusion, the NRC staff finds that although the guidance of NEI 12-06, Revision 0 has not been met, if these alternatives are implemented as described by the licensee, they will meet the requirements of the order.

3.15 Conclusions for Order EA-12-049

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

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 23], the licensee submitted an OIP for CNS in response to Order EA-12-051. By letter dated June 24, 2013 [Reference 24], the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response by letter dated July 23, 2013 [Reference 25]. By letter dated October 28, 2013 [Reference 26], the NRC staff issued an Interim Staff Evaluation (ISE) and RAI to the licensee.

By letters dated August 26, 2013 [Reference 27], February 26, 2014 [Reference 28], August 14, 2014 [Reference 29], February 16, 2015 [Reference 30], and August 3, 2015 [Reference 31], the licensee submitted status reports for the Integrated Plan. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable SFP instrumentation (SFPI) which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letters dated May 1, 2015 [Reference 39], and February 15, 2016 [Reference 19], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved at CNS, Units 2 and 1 respectively. By letter dated March 31, 2016 [Reference 40], the licensee provided supplemental information regarding Order EA-12-051 compliance.

The licensee has installed a SFPLI system designed by AREVA Americas, Inc. (AREVA). The NRC staff performed a generic audit review of the vendor's SFPLI system design specifications, calculations and analyses, test plans, and test reports after the submittal of licensees' OIPs in response to Order EA-12-051. The staff issued an audit report documenting the results of this audit on September 15, 2014 [Reference 32].

The NRC staff also performed an onsite audit at CNS to review site-specific aspects of the implementation of Order EA-12-051. The scope of this audit included a review of the licensee's plans and confirmation of whether: (a) the site's seismic and environmental conditions are enveloped by the equipment qualifications, (b) the planned equipment installation will meet the requirements and vendor's recommendations, and (c) the planned program features will be in accordance with the Order EA-12-051 requirements. By letter dated February 20, 2015 [Reference 18], the NRC issued an audit report on the licensee's progress toward meeting the requirements of Order EA-12-051.

4.1 Levels of Required Monitoring

In its OIP, the licensee stated that indication of SFP level will be provided from approximately normal pool water level (Elevation 598'-6") down to approximately the top of irradiated fuel assemblies seated in the storage racks (Elevation 573'). According to the licensee, Level 1 indicates water level greater than the point at which pump suction is presumed to be lost (Elevation 597' - 4"). Level 2 indicates water level greater than 10' above the highest point of any fuel racks, or greater than Elevation 583'. This monitoring level ensures there is adequate water level to provide substantial radiation shielding for personnel responding to beyond-design-basis external events and to initiate SFP makeup strategies. Level 3 indicates water level less than 12 inches above the highest point of any fuel storage rack. This monitoring level ensures there is adequate water level above the stored fuel seated in the storage racks.

By letter dated July 23, 2013, the licensee changed the Level 1 Elevation to 597'-6". This was based on a determination as follows: (1) the normal SFP water level is 598'-6" elevation, (2) the SFP cooling pump suction piping submergence is lost when water level decreases below 597'-4" elevation, (3) abnormal procedures secure the SFP cooling pump when water level decreases to 1' below normal. By letter dated July 23, 2013, the licensee also provided a figure depicting the proposed Level 1, 2, and 3. In this figure, Level 1 is designated as 597'-6" elevation, Level 2 is designated as 583'-0" elevation, and Level 3 is designated as 573'-0" elevation. This figure also shows the SFP level measurement range of 573'-0" elevation to 599'-0" elevation.

The NRC staff notes that Level 1 designation is adequate for normal SFP cooling system operation and it is also adequate to ensure the required SFP cooling pump net positive suction head. This level also represents the higher of the two points described in NEI 12-02 for Level 1. The Level 2 designation uses the first of the two options described in NEI 12-02 for Level 2, which is approximately 10 feet above the top of the fuel rack. The Level 3 designation on the licensee's figure corresponds to the top of the fuel rack.

Based upon the licensee's descriptions the NRC staff finds that the proposed Levels 1, 2, and 3 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 requires that the SFPLI shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Below is the staff's assessment of the design features of the SFP level instrumentation.

4.2.1 Design Features: Instruments

By letter dated July 23, 2013, the licensee provided a description of the proposed SFP level instrumentation: the primary SFP level channel will utilize wave guided radar technology, which will have a wave guided pipe and receiving horn located in the SFP area. The wave guided pipe and horn will contain no organic materials and will not be susceptible to degradation due to exposure to radiation, heat, or steam. The associated primary channel electronics will be remotely located from the SFP inside the Seismic Category I Auxiliary Building. This channel will have battery backup capacity and will provide remote control room level monitoring capability. The backup SFP level channel will consist of a pressure transmitter that senses SFP head (level) based on a process connection to the SFP transfer tube. The backup level instrument readout will be located in the MCR. The backup level instrument will be powered from the unit vital batteries.

The NRC staff finds that the licensee's design, with respect to the number of channels 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 two SFP level instrument channels will be installed in spatially separated locations and arranged in a manner to reduce the potential for common damage to both channels. Each channel will be installed within or adjacent to the Spent Fuel Buildings on each unit, which are Seismic Category I structures capable of withstanding missiles and other external events.

By letter dated March 31, 2016, the licensee stated that Drawings CN-1200-11.02 (Unit 1) and CN-1200-12.02 (Unit 2) show features of the Spent Fuel Buildings at Elevation 605 + 10, including the layout/dimensions of the Spent Fuel Pools. Drawings CNM 1336.04-0005 001 (Unit 1) and CNM 2336.04-0005 001 (Unit 2) show the waveguide pipe layout from the horn above the pools to the sensor. Drawings CN-1210-31 (Unit 1) and CN-1210-30 (Unit 2) show the guard assemblies for the piping, and also shows the piping relative to the corners of the pools. Drawing CN-1(2)499-01.10-00 shows the location of the backup transmitters. Drawings CN-1(2)499-01.07-00 and CN-1(2)878-01 show the location of the primary transmitter and control panel. Since the cabling associated with the primary channel originates from Elevation 594 of the Auxiliary Building and the cabling associated with the backup channel originates from Elevation 557 for Unit 1 and Elevation 556 for Unit 2 (CN-1(2)499-01.10-00) of the Reactor Building Annulus, the cabling is separated.

The NRC staff reviewed the above drawings and verified them during the onsite audit walk down. The staff noted that the primary instrument channel's sensor is located at the northwest corner of the SFP for Unit 1 and southwest corner of the SFP for Unit 2. The Unit 1 backup channel's pressure transmitter is located at the at the 557 foot elevation of the Unit 1 Reactor Building Annulus area and the Unit 2 backup channel's pressure transmitter was located at the at the 556 foot elevation of the Unit 1 Reactor Building Annulus area. For each unit, the primary instrument channel's sensor and backup channel's transmitter are located in different buildings and elevation.

The licensee further provided information on the sensing line tubing design for the backup instrument channel's pressure transmitter by letter dated March 31, 2016: the backup monitor is classified as nuclear safety-related and the sensing line tubing is also classified as nuclear safety related. The tubing is about 35 feet in length and is sloped downward a minimum of ¼" per foot from the process tap. Tubing is installed in accordance with all applicable nuclear safety related guidance. An analysis of ELAP temperatures in areas adjacent to the annulus shows that the lower annulus area should not be subject to freezing during cold outside temperatures. There are a number of transmitters associated with different plant systems currently located in the lower annulus area. Most of these devices are routinely accessed for calibration and maintenance. The pressure transmitter for SFP level will also be accessible, as needed.

The NRC staff notes that the licensee adequately addressed the SFP level instrument arrangement requirements by providing reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. For the cabling routing of the SFP level instrument, the NRC staff reviewed the following drawings and verified them during the walk down:

- CN-2710-02.04-03, "Auxiliary Building Cable Trough Geometry, Below EL. 594' + 0"," Revision 2
- CN-2710-02.11-06, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 587' + 0" Computer Cable Routing," Revision 16
- CN-2710-02.11-02, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 591' + 0" Computer Cable Routing," Revision 14
- CN-2710-02.11-03, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 590' + 0" Computer Cable Routing," Revision 16
- CN-2894-01.01, "Computer Cable Routing Auxiliary Building Plan Below EL. 611' + 0" Cols. EE-QQ & 57-64," Revision 23
- CN-2893-01.01, "Computer Cable Routing Auxiliary Building Plan Below EL. 594' + 0" Cols. EE-QQ & 57-64," Revision 27
- CN-2710-02.11-08, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 585' + 0" Computer Cable Routing," Revision 10
- CN-2710-02.11-11, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 582' + 0" Computer Cable Routing," Revision 5
- CN-2710-02.04-08, "Auxiliary Building Cable Trough Geometry, Below EL. 594' + 0"," Revision 3
- CN-2710-02.11-07, "Auxiliary Building Electrical Equipment Layout Cable Room, EL. 586' + 0" Computer Cable Routing," Revision 20
- CN-2897-01.01, "Computer Cable Routing Auxiliary Building Elect. Pene. & Swgr. Room Plan Below EL. 594' + 0"," Revision 38
- CN-2918-01.01, "Computer Cable Routing Reactor Building Annulus Elevation Azimuth 180° – 0°," Revision 27

Based on that review the NRC staff determined that there is sufficient channel separation within the SFP area between the primary and backup level instrument channels and routing cables to provide reasonable protection against loss of indication of SFP level due to missiles that may result from damage to the structure over the SFP.

Based on the discussion above, 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 <u>Design Features: Mounting</u>

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

By letter dated March 31, 2016, the licensee further stated that the primary channel is mounted seismically. The mounting designs for the electronic sensor support, horn support, and intermediate supports were qualified considering the total weight of the waveguide piping and its components and the seismic accelerations for the building structure. To meet the design criteria for a beyond-design-basis (BDB) Event, the loading for the mounting supports were generated using a minimum of two times the SSE accelerations. The electronic sensor mounting support is qualified by a generic calculation using a simple C-channel steel section that is welded centrally on a ½ " thick steel base plate on the Auxiliary Building concrete wall. This Auxiliary Building wall is classified as Seismic Category I. The base plate is anchored to the wall with four (4) concrete anchor bolts. The generic sensor mounting support was designed for generic enveloping seismic accelerations of 10g (horizontal) and 6.67g (vertical), which readily envelopes the site specific response spectra. This support is qualified to withstand deadweight, seismic and sloshing loads, as well as the loads acting on it from the horn end assembly. A hydrodynamic SFP analysis for CNS showed that no sloshing loads apply to the CNS horn installation. Liquid sloshing in response to the required response spectrum is small such that inventory does not spill from the SFP nor reach a height that interacts with the SFP radar horn.

The licensee's letter dated March 31, 2016, states that Calculation CNC-1139.14-08-0001 provides the design of the intermediate waveguide mounting supports. The piping which runs horizontally above the floor is protected by a guard assembly which also incorporates supports. All of the mounting supports for the waveguide piping will be attached to either the concrete wall or concrete floor. These are Seismic Category I concrete structures with a minimum concrete strength of 5000 psi. The mounting design for the power control panel is qualified considering the total weight of the panel and its associated components and the seismic accelerations for the building structure. To meet the design criteria for a BDB Event, the loading for the panel was generated using a minimum of two times the SSE seismic accelerations. The backup channel pressure transmitter is mounted on the inside wall of the Reactor Building in the lower annulus. It is mounted in accordance with the licensee's standard practice for installation of a safety related transmitter.

The NRC staff review notes that the mounting attachments were designed in a conservative manner with respect to the maximum seismic ground motion in the design basis of the plant (twice the SSE). The staff finds that the licensee adequately incorporated the design criteria and used an appropriate methodology 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. The structural integrity of the affected structures was also adequately addressed. The staff reviewed the following documents to support this conclusion:

- Drawing 02-9214604D, "Catawba Nuclear Station Unit 1 & 2 Vegapuls 62 ER Antenna Wave Guide Mount Assembly," Revision 0

- Calculation 32-9221238-000, "Qualification for a Waveguide Type "B" Support and Horn End assembly for AREVA Spent Fuel Pool Level Monitoring Instrumentation," Revision 18, dated January 30, 2014

- CNC-1336.04-00-0001, "Seismic Induced Hydrodynamic Response in the Catawba and McGuire Spent Fuel Pools (Sloshing Analyses)," Revision 0

Based on the licensee's submittals, as confirmed during the site audit, the NRC staff finds the licensee's proposed mounting design is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

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

In its OIP, the licensee stated that augmented quality provisions will be applied to ensure the rigor of the qualification documentation reviews and in-plant modification installation oversight is sufficient to ensure compliance with the qualification requirements above.

The NRC staff notes that 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 SFP level instrumentation's qualification and testing during the vendor audit for temperature, humidity, radiation, shock and vibration, and seismic [Reference 32]. The staff further reviewed the anticipated CNS's environmental conditions during the on-site audit [Reference 18].

4.2.4.2.1 Temperature, Humidity, and Radiation

By letter dated March 31, 2016, the licensee stated that formal environmental analyses were performed for limited areas, including the Auxiliary Building SFP Level Monitoring Area, during a postulated ELAP event. Results indicate temperature for area of the Auxiliary Building on Elevation 594 will not exceed 120°F after 7 days without mitigating cooling actions. The limiting ambient design temperature for the primary channel sensor and power control panel components is 149°F, and is acceptable for the proposed location. Exposure to saturated steam conditions present in the SFP area will be minimal. The horn assembly over the SFP has a sealed glass cover to prevent steam or high humidity air from entering the waveguide pipe. The ambient humidity in the Auxiliary Building monitoring location is expected to be below 100 percent relative humidity during a postulated ELAP event. The sensor has been tested by a protocol that varies the room temperature from normal 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 enclosure provides protection to the internal components from the effects of high humidity environments. The backup instrument channel's transmitters are safety-related and qualified for the normal and BDBEE environment.

By letter dated March 31, 2016, the licensee stated that a CNS specific radiation dose calculation was performed. This calculation modeled a predicted dose associated with the pool remaining at Level 3 for 7 days. According to the licensee, "normal dose" in the radiation zone of the transmitter is determined as 130 Rads per year (5.2E3 Rads/40 years). The predicted seven day event dose is 233 Rads. This calculation indicates the sensitive electronic components may need to be replaced in less than 6 years of service life. The licensee also stated that this calculation is conservative and radiation dose at the equipment mounting location may be collected over a period of time and the calculation revised, if appropriate.

The NRC staff reviewed the licensee's submittal, as well as supporting documentation, and finds that the equipment qualification envelops the expected CNS radiation, temperature, and humidity conditions during a BDB event such that the SFP instrumentation should maintain its functionality during the expected BDB conditions.

4.2.4.2.2 Shock and Vibration

By letter dated February 15, 2016, the licensee stated that the sensor, displays, and power control panel have been tested and/or analyzed for shock and vibration in accordance with International Electrotechnical Commission (IEC) Standards, IEC 60068-2-6 (vibration) and IEC 60068-2-27 (shock). The test parameter values specified envelope the expected level for the equipment installed location. The vibration testing deviated from the IEC 60068-2-6 recommended frequency range and displacement magnitude for large power plant equipment. In-lieu of the 10-55 Hz and minimum displacement of 0.15 millimeter (mm), the power and control panel vibration testing utilized a narrower frequency band (5-25 Hz) and a more limiting displacement magnitude (1.6 mm). These values were deemed to be acceptable and enveloping for equipment rigidly mounted to the Seismic Category I structures, based on the licensee's engineering judgment. The shock testing deviated from the IEC 60068-2-27 recommended peak acceleration and duration for land-based permanently installed equipment. In-lieu of the 15 g's peak acceleration and duration of 11 milliseconds (m-sec), the power and control panel vibration testing utilized and acceleration of 10g with a 6 m-sec duration. These values were deemed to be acceptable and enveloping for equipment rigidly mounted to a seismic Category I structure, based on the licensee's engineering judgment. Testing and analyses of the horn cover and adhesive support the components can tolerate horizontal and vertical accelerations up to 100g and SFP sloshing loads up to 3.37 psi.

The NRC staff reviewed the licensee's submittal and finds that the test parameters envelope the CNS expected shock and vibration conditions during a postulated BDBEE. Therefore the staff finds that the licensee adequately addressed the equipment reliability of the SFPLI with respect to shock and vibration.

4.2.4.2.3 Seismic

In its OIP, the licensee stated, in part, that the level instrumentation is to be designed to remain functional following a Safe Shutdown Earthquake. By letter dated March 31, 2016, the licensee further stated that the response spectra used for seismic testing of the SFP primary level instrumentation and the electronics units envelop the CNS design-basis seismic spectra for the locations where the equipment is installed. The seismic testing and analysis performed is in accordance with Institute of Electrical and Electronics Engineers (IEEE) standard IEEE 344-2004, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Sites" methodology per site procedures. The licensee performed a calculation which shows that liquid sloshing in response to the required response spectrum is small such that inventory does not spill from the SFP or reach a height that interacts with the SFP radar horn. The licensee evaluated waveguide and horn end assembly for deadweight and seismic loads. All of the components are attached to reinforced concrete structural members. According to the licensee's site support specification, new attachments to concrete structural members are acceptable if the reaction loads and moments are below a particular threshold. All the new components have reaction loads and moments below the specified threshold. Therefore the structures to which the components are attached are qualified.

During the onsite audit, the NRC staff reviewed calculation CNC-1336.04-00-0001, "Seismic Induced Hydrodynamic Response in the CNS and McGuire Spent Fuel Pools," Revision 0. The scope of calculation CNC-1336.04-00-0001 is to evaluate the hydrodynamic response of the SFP to a safe shutdown earthquake (design-basis earthquake) response spectrum. The staff noted that the calculation used the GOTHIC 8.0 thermal-hydraulic analysis software package, which includes adjustable body force terms in the momentum equation for each coordinate direction. The artificial seismic acceleration time histories were generated from the damped ground motion response spectra compliant with the requirements of IEEE Std. 344-2004 and NUREG-0800, Section 3.7.1. According to CNC-1336.04-00-0001, the inventory in the SFP is excited in response to seismic motion of the pool structure. The SFP hydrodynamic response includes pool sloshing, wave formation, bulk and convective information. Pool sloshing and wave formation may reach the installed instrument elevation resulting in hydrodynamic impact and drag forces on the instrument structure. The calculation evaluated the flow conditions local to the guided wave instrument to determine whether the SFP inventory interacts with instrument resulting in hydrodynamic loads. In its conclusion, the calculation stated that there are no reportable hydrodynamic loads. The analyzed maximum slosh heights are 0.94 feet for the original acceleration input and 1.31 feet for the case with acceleration input rotated by 10 degrees from the base orientation, both of which are well below the distance between the SFP liquid surface and the horn (6 feet).

The NRC staff found that the assumptions and methodology used in the sloshing analysis for the radar horn, waveguide, and horn end assembly were adequate. However, during the onsite audit, the staff had concerns regarding the lack of information on the seismic qualification of the Rosemount Model 3154N differential pressure transmitter, Acopian Model VB35GT10 power supply, and Weschler Model VX-252 panel indicator. In response to the staff's concerns, in its letter dated March 31, 2016, the licensee provided the seismic testing results for the Rosemount Model 3154N differential pressure transmitter, the summary of a test report for Acopian Model VB35GT10 power supply, and the seismic testing summary for the Weschler Model VX-252 panel indicator. The NRC staff noted that the SFPLI equipment were tested to the seismic conditions that envelope the CNS's expected highest SSE.

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

4.2.5 Design Features: Independence

In its OIP, the licensee stated that the two SFP level instrument channels for each pool will be physically and electrically independent of one another. The associated cabling, power supplies and indication for each level instrument channel will be routed separately from each other.

By letter dated March 31, 2016, the licensee further stated that the primary and backup SFP level channels employ diverse sensing technology. The primary and backup channels are independent in terms of widely separated physical locations. The two channels utilize no sharing of power sources, wiring, enclosures, electronics or temperature environments, with the exception of the indicators in the control room. The two channels utilize the same type of proven control board indicator on a common control board, but each is separate and independent of the other.

Based on the review of the licensee's submittals, supplemented by walk down verification, the NRC staff notes that with the licensee's proposed design, the loss of one level measurement channel would not affect the operation of the other independent channel under BDB event conditions. The acceptability of the instrument channels' physical separation is discussed in Subsection 4.2.2, "Design Features: Arrangement". Therefore, the staff finds the licensee's proposed design, with respect to instrument channel independence, is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.6 Design Features: Power Supplies

In its OIP, the licensee stated that the two instrument channels for each unit will be powered normally by separate power supplies backed up by rechargeable or replaceable batteries. The backup power sources will have sufficient capacity to maintain the level indication function until offsite power or other emergency resource availability is reasonably assured.

By letter dated March 31, 2016, the licensee further stated that the primary wide range Spent Fuel Pool Level power control panel normally operates on 120 Vac power which is converted internally to 24 Vdc. This power control panel is also provided with a 24 Vdc bank of internal batteries. The monitor automatically switches to the internal batteries upon loss of the normal 120 Vac source. For the expected minimum temperature at the Power Control Panel location the batteries are rated to last a minimum of 7 days. The backup channel receives 120 Vac power from a safety-related inverter panel board which is fed from the vital batteries of that unit. A FLEX strategy restores charging to the vital battery chargers during Phase II. Power will continue to be available to the backup channel through the vital inverters for as long as needed. The vital batteries providing power for the backup channel will carry all loads not stripped for a minimum of 9 hours. By that time, battery charging will be restored through FLEX power and the backup level channel will see no interruption of power. Post-modification testing was also performed with the results indicating the primary radar instrument Unit 1 batteries were capable of providing SFP level indication for 450.72 hours (18.78 days) and the Unit 2 batteries were capable of providing SFP level indication for 429 hours (17.875 days).

The NRC staff finds the licensee's proposed power supply design provides sufficient battery capability to power the instrumentation until the batteries can be supplemented with power being provided as part of the licensee's BDBEE mitigating strategy. The staff thus concludes that the licensee's power supply for SFPI is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.7 Design Features: Accuracy

In its OIP, the licensee stated that the new SFPLI will be designed to maintain their design accuracy without recalibration following a power interruption or change in power source. Additionally, instrument accuracy will be designed to allow trained personnel to determine when the actual level exceeds the specified lower level of each indicating range without conflicting or ambiguous indication.

By letter dated March 31, 2016, the licensee further stated that the normal and BDB instrument loop uncertainties for the primary and backup instrument channels have been calculated. The total loop uncertainties for the primary instrument loop during normal operation is +/- 9.5 inches, and the total loop uncertainties for the backup instrument loop during normal operation is approximately +/- 15 inches. The total loop uncertainties for the primary instrument loop during ELAP conditions is +/- 10.91 inches, and the total loop uncertainties for the backup instrument loop during ELAP is +/- 11.38 inches. For the backup loop the process measurement allowance ranges from approximately +21 inches to -15 inches. The licensee determined the required scaling, calibration tolerances, and channel check allowances based on the uncertainty calculation. The methodology used for determining the maximum allowed deviation between the level indications is the square root sum of the squares of the loop uncertainties of the two compared channels. Channel checks are performed at least every 31 days in accordance with plant procedures, as required by Selected Licensee Commitment (SLC) 16.7-17. If the compared channels deviate by more than the allowed level difference, the mismatch will be investigated and resolved.

The NRC staff noted that the manufacturer's reference accuracy for the primary SFP level channel is ±1 inch for normal conditions and ± 3 inch for BDB conditions (including 212°F saturated steam). The VEGAPULS 62 ER SFPI is designed to maintain its accuracy after a power interruption. AREVA performed testing on VEGAPULS 62 ER to verify channel accuracy after changing power sources without recalibration. Each instrument measured 10 different locations including the maximum and minimum locations of the instrument span. The instrument accuracy was within ±1 inch of the actual target location for all measurements taken. AREVA then tested the instrument accuracy of the two test units while performing a horn rotation test, and found the readings taken to be within ±1 in. Finally, separate steam and fluid drain tests were performed and the instrument readings stayed within ±3 in. of the original reading (under normal conditions). As part of the onsite audit activity, the staff reviewed the licensee's calculation CNC-1210.04-00-0136, "Wide Range Spent Fuel Pool Level Scaling and Accuracy Calculation," Revision 1. The staff noted that the calculated uncertainties for the SFP level instrument are adequate to determine the calibration span and channel check tolerances. The uncertainties the calculation accounted for include: the device accuracy/drift, temperature effect, measuring and test equipment effect, calibration effect, and boric acid concentration effect. The staff found the methodology, assumptions, and design inputs the licensee used in the calculation acceptable.

The NRC staff found the licensee adequately addressed the SFPLI accuracy requirements including the expected instrument channel accuracy performance under both normal and BDB conditions. The staff concludes that, if implemented properly, the instrument channels will maintain the designed accuracy following a power source change or interruption without the need of recalibration.

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

4.2.8 Design Features: Testing

In its letter dated March 31, 2016, the licensee stated that for the primary instrument channel, the expected calibration drift is negligible associated with the through air radar transmitter, and all that is required is to verify that the system is functioning correctly. A minimum two point functional check is performed for the primary channel. This will be is achieved by varying pool water level over a small range adequate to verify the system functionality at different water levels. In addition, a six point calibration is performed on the control room indicator. The backup channel consists of a differential pressure transmitter, which is provided with a test tee connection. The test tee allows a pressure source to be connected to the transmitter and perform a six point string calibration through to the control room indicator per plant procedure. All indicators for the wide range primary and backup channels, as well as the existing narrow range level signal, are available in the MCR for channel checks. Channel checks are performed by verifying the primary and backup channels agree within a specified tolerance at least every 31 days. Channel calibrations are required within 60 days prior to the start of a planned refueling outage. The primary channel vendor recommends periodic checks to verify proper operation of the battery backup capability. The vendor also recommends that the batteries be replaced at each calibration period. During this calibration surveillance prior to each refueling outage the swap over from ac to dc power will be tested and the batteries will be replaced prior to completion of the calibration.

The NRC staff noted that the SFPLI is adequately designed to provide the capability for routine testing and calibration including in-situ testing/calibration. By comparing the levels in the instrument channels and the maximum level allowed deviation for the instrument channel design accuracy, the operators could determine if recalibration or troubleshooting is needed. The staff finds the licensee's proposed SFPI design allows for testing consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.9 Design Features: Display

In its letter dated March 31, 2016, the licensee stated that the primary operator displays for all primary and back up wide range SFPLI channels are mounted in the MCR.

The NRC staff notes that if implement 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 following an ELAP event is considered acceptable. Therefore, the staff finds that the licensee's proposed location and design of the SFPI displays is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3 Evaluation of Programmatic Controls

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

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that personnel shall be trained in the use and the provision of alternate power to each instrument channel. Station personnel performing functions associated with the SFPLI will be trained to perform the job specific functions necessary for their assigned tasks. The SAT will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training.

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

4.3.2 Programmatic Controls: Procedures

In its OIP, the licensee stated that procedures will be developed using guidelines and vendor instructions to address the maintenance, operation and abnormal response issues associated with the SFP level instrumentation. Procedures will also address the strategy to ensure SFP water addition is initiated at an appropriate time consistent with implementation of NEI 12-06, Diverse and FLEX Implementation Guide. In its letter dated March 31, 2016, the licensee provided a list of CNS procedures related to the SFPLI.

In addition to the procedure list, the licensee stated that emergency procedures, abnormal procedures and FLEX guidelines provide guidance for assessing any ELAP impact on the SFP and they also specify actions for mitigation. The licensee's letter indicates that an operations surveillance procedure verifies compliance with channel check surveillance items. The maintenance procedures perform periodic calibration of the primary and backup SFPLI. Specifically, they perform a functional check of primary channel battery backup capability, verify proper operation of the level instrumentation and provide instruction for equipment calibration and adjustment within design accuracy requirements.

The NRC staff notes that the licensee has established procedures for the testing, surveillance, calibration, and operation of the primary and backup SFP level instrument channels. Therefore, the staff finds that the licensee's proposed procedures are consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

By letter dated March 31, 2016, the licensee stated that programmatic controls have been established to ensure the performance of periodic channel checks, functional tests, calibration, and maintenance for the instrument channels. The programmatic controls have been established by a new SLC 16.7-17. The new SLC specifies the required frequency of performance for periodic channel checks and functional checks, as appropriate. The SLC outlines allowed out of service time frames consistent with NEI 12-02 requirements. The SLC specifies required remedial actions, in the event one or more channels cannot be restored within the allowed out of service time-frame. The remedial actions are consistent with NEI 12-02 requirements. The SLC further requires functional testing be performed to verify proper channel operation within 60 days of a planned refueling outage, as required by NEI 12-02. The calibration frequency is controlled by the SLC and maintained by the plant preventive maintenance program. The frequency is based on manufacturer recommendations and/or operating experience. The CAP would formally evaluate "functionality" for the SFP level channels and establish appropriate compensatory measures. The CAP would further establish appropriate procedural and process controls to ensure performance of any required compensatory measures.

The maintenance procedures will not explicitly address any expedited or compensatory actions for a channel that is not restored to functional within 90 days. As required by NEI 12-02, compensatory actions must be implemented if one channel is not expected to be restored to functional within 90 days. The remedial actions listed in the SLC Bases and the CAP will evaluate and establish appropriate compensatory actions for a channel that cannot be restored to functional status within 90 days. In the letter, the licensee also provided a list of potential compensatory actions as following:

SLC Condition B (one channel out-of-service for greater than 90 days) potential remedial actions:

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

SLC Condition C (two channels out of service) potential remedial actions:

- Increased operator visual surveillance of the SFP level and area
- Maintenance of elevated SFP level
- Reduction of SFP temperature

- Supplemental operations staffing
- Pre-staging of FLEX support equipment (nozzles, hoses, etc.) which are relied upon for SFP makeup (pre-staged equipment would be located within Seismic Category I structures)

The NRC staff notes that the necessary testing and calibration of the primary and backup SFP level instrument channels has been established to maintain the channels at their design accuracy. Further the staff notes that the testing and calibration are consistent with the vendor recommendations and the proposed compensatory actions for instrument channel(s) out-of-service appear to be consistent with guidance in NEI 12-02.

4.4 Conclusions for Order EA-12-051

In its letters dated February 15, 2016, and March 31, 2016, 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 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 CNS according to the licensee's proposed design, it should adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on these two orders. The staff completed an onsite audit in October 2014 [Reference 18]. The licensee reached its final compliance date on December 16, 2015, and has declared that both of the reactors are in compliance with the orders. The purpose of this SE is to document the strategies and implementation features that the licensee has committed to and which NRC staff has evaluated to be satisfactory for compliance with these orders. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs which if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051.

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," 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," February 17, 2012 (ADAMS Accession No. ML12039A103)

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

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

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

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

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

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

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

10. Duke letter to NRC, "Catawba Nuclear Station (CNS), Units 1 and 2, Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, EA-12-049," February 28, 2013 (ADAMS Accession No. ML13066A173)

11. Duke letter to NRC, "Catawba, Units 1 & 2, First Six-Month Status Report in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order No. EA-12-049)," August 28, 2013 (ADAMS Accession No. ML13298A010)

12. Duke letter to NRC, "Second Six-Month Status Report in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," February 28, 2014 (ADAMS Accession No. ML14065A038)

13. Duke letter to NRC, "Catawba, Units 1 and 2, Third Six-Month Status Report in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," August 28, 2014 (ADAMS Accession No. ML14247A232)

14. Duke letter to NRC, "Catawba, Units 1 and 2 - Fourth Six-Month Status Report in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," February 26, 2015 (ADAMS Accession No. ML15061A124)

15. Duke letter to NRC, "Catawba, Units 1 and 2 - Fifth Six-Month Status Report in Response to March 12, 2012, Commission Order to Modify Licenses With Regard To Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," August 26, 2015 (ADAMS Accession No. ML15240A066)

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

17. Letter from Jeremy S. Bowen (NRC) to Kelvin Henderson (Vice President – Catawba Nuclear Station), "Catawba Nuclear Station, Units 1 and 2 – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigation Strategies) (TAC Nos. MF1162 and MF1163)," February 6, 2014 (ADAMS Accession No. ML13364A175)

18. Letter from Peter Bamford (NRC) to Kelvin Henderson (Vice President – Catawba Nuclear Station), "Catawba Nuclear Station, Units 1 and 2 – Report for the Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051, (TAC Nos. MF1162, MF1163, MF1060, MF1061)," February 20, 2015 (ADAMS Accession No. ML15035A679)

19. Duke letter to NRC, "Catawba, Units 1 & 2, Final 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 to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation,' for Catawba Nuclear Station," February 15, 2016 (ADAMS Accession No. ML16049A041)

20. U.S. Nuclear Regulatory Commission, "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)

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Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," August 14, 2014 (ADAMS Accession No. ML14227A717)

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34. Letter from William Dean (NRC) to Power Reactor Licensees, "Coordination of Requests for Information Regarding Flooding Hazard Reevaluations and Mitigating Strategies for Beyond-Design Bases External Events," September 1, 2015 (ADAMS Accession No. ML15174A257).

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

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41. Letter from Jack R. Davis (NRC) to Jack Stringfellow (PWROG), regarding Boron Mixing, dated January 8, 2014 (ADAMS Accession No. ML13276A183)

42. Letter from Nicolas Pappas (NEI) to Jack R. Davis (NRC), "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," dated August 27, 2013 (ADAMS Accession No. ML13241A186)

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

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50. Duke letter to NRC, "Catawba, Units 1 & 2, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 10 CFR 50.54(f) RAI Per 10 CFR 50.54(f) re Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 31, 2014 (ADAMS Accession No. ML14093A052)

51. Letter from Frankie Vega (NRC) to Kelvin Henderson (Site Vice President – Catawba Nuclear Station), "Catawba Nuclear Station, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Revaluations Relating to Recommendation 2.1 of the NTTF Review," April 27, 2015 (ADAMS Accession No. ML15096A513)

52. Letter from Andrew Mauer (NEI) to Jack R. Davis (NRC), "Nuclear Energy Institute (NEI), Request for Endorsement of Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation (EPRI 3002007148)," February 23, 2016 (ADAMS Accession No. ML16055A017)

53. Letter from William M. Dean (NRC) to Applicable Power Reactor Licensees, "Final Determination Of Licensee Seismic Probabilistic Risk Assessments Under The Request For Information Pursuant To Title 10 Of The Code Of Federal Regulations 50.54(F) Regarding Recommendation 2.1 'Seismic' of Near-Term Task Force Review of Insights From the Fukushima Dai-ichi Accident," October 27, 2015 (ADAMS Accession No. ML15194A015)

54. Letter from Nicolas DiFrancesco (NRC) to Kelvin Henderson (Site Vice President – Catawba Nuclear Station), "Catawba Nuclear Station, Units 1 And 2 - Staff Review Of Interim Evaluation Associated With Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1," March 17, 2016, (ADAMS Accession No. ML16072A037)

55. Duke letter to NRC, "Catawba Nuclear Station, Units 1 and 2, Spent Fuel Pool Evaluation Supplemental Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 20, 2016 (ADAMS Accession No. ML16204A060)

56. NUREG/CR-7188,"Testing to Evaluate Extended Battery Operation in Nuclear Power Plants," May 2015 (ADAMS Accession No. ML15148A418)

57. Letter from Frankie Vega (NRC) to Kelvin Henderson (Site Vice President – Catawba Nuclear Station), "Catawba Nuclear Station, Units 1 and 2 - Staff Review of Spent Fuel Pool Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1," August 11, 2016 (ADAMS Accession No. ML16222A368)

58. Duke letter to NRC, Catawba, Units 1 & 2, "Supplemental Information to Final 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 to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation," September 27, 2016 (ADAMS Accession No. ML16273A303)

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Date: October 20, 2016

By letter dated February 28, 2013 (ADAMS Accession No. ML13086A095), Duke submitted its OIP for CNS in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached SE. By letters dated October 28, 2013 (ADAMS Accession No. ML13281A562), and February 20, 2015 (ADAMS Accession No. ML15035A679), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated May 1, 2015 (ADAMS Accession Nos. ML15126A277), Duke submitted a compliance letter for CNS, Unit 2, in response to Order EA-12-051. By letter dated February 15, 2016 (ADAMS Accession No ML16049A041), Duke submitted a compliance letter for CNS, Unit 1, in response to Order EA-12-051. The compliance letters stated that the licensee had achieved full compliance with Order EA-12-051 for each unit, as applicable. By letter dated March 31, 2016 (ADAMS Accession No. ML16095A208), the licensee submitted supplemental information regarding compliance with Order EA-12-051 at CNS.

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

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

Sincerely,

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

/RA/

Docket Nos.: 50-413 and 50-414
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