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

June 20, 2016

Mr. Steven D. Capps
Vice President - McGuire Site
Duke Energy Carolinas, LLC
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE 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. MF1160, MF1161, MF1062, AND MF1063)

Dear Mr. Capps:

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. ML13063A185), Duke Energy Carolinas, LLC (Duke, the licensee) submitted its OIP for McGuire Nuclear Station, Units 1 and 2 (MNS) 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. 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 January 16, 2014 (ADAMS Accession No. ML13338A406), and October 9, 2014 (ADAMS Accession No. ML14241A454), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated December 7, 2015 (ADAMS Accession No. ML15343A010), Duke submitted its compliance letter and the Final Integrated Plan (FIP) in response to Order EA-12-049. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-049.

By letter dated February 28, 2013 (ADAMS Accession No. ML13086A095), Duke submitted its OIP for MNS in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051.

These reports were required by the order, and are listed in the attached safety evaluation. By 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 letters dated October 28, 2013 (ADAMS Accession No. ML13281A791), and October 9, 2014 (ADAMS Accession No. ML14241A454), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated December 7, 2015 (ADAMS Accession No. ML15343A010), Duke submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Duke's strategies for MNS. The intent of the safety evaluation is to inform Duke on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML14273A444). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact John Boska, Orders Management Branch, McGuire Nuclear Station Project Manager, at 301-415-2901 or at John.Boska@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.: 50-369 and 50-370

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

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

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

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

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC 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 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 an Overall Integrated Plan (OIP) for the McGuire Nuclear Station, Units 1 and 2 (MNS, McGuire) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 27, 2014 [Reference 12], August 27, 2014 [Reference 13], February 26, 2015 [Reference 14], and August 26, 2015 [Reference 45], the licensee submitted six-month updates to the OIP. By letter dated August 28, 2013 [Reference 15], the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 34]. By letters dated January 16, 2014 [Reference 16] and October 9, 2014 [Reference 17], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated December 7, 2015 [Reference 18], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP). By letter dated May 5, 2016, the licensee submitted a supplement to the compliance letter [Reference 54].

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 ultimate heat sink. Thus, the ELAP with loss of normal access to the ultimate heat sink 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 MNS, Units 1 and 2, are four loop Westinghouse pressurized-water reactors (PWRs) with ice condenser containments. The FIP describes the licensee's three-phase approach to mitigate a postulated ELAP event.

At the onset of an ELAP, both reactors are assumed to trip from full power. The reactor coolant pumps coast down, and flow in the reactor coolant system (RCS) transitions to natural circulation. Operators will take prompt actions to minimize RCS inventory losses by isolating potential letdown paths. Decay heat is removed by steaming to atmosphere from the steam generators (SGs) through the SG power-operated relief valves (PORVs) or main steam safety valves, and makeup to the SGs is initially provided by the turbine-driven auxiliary feedwater (TDAFW) pump taking suction from the auxiliary feedwater storage tank (CAST). However, the CAST is not protected from all hazards. If it is not available, the TDAFW pump suction will automatically swap to take water from the buried condenser circulating water piping, which utilizes lake water but is fully protected from all applicable hazards. Subsequently, operators would begin a controlled cooldown and depressurization of the RCS by manually operating the SG PORVs. The SGs would be depressurized in a controlled manner to about 290 pounds per square inch gage (psig) over a period of several hours, and then maintained at this pressure while the operators borate the RCS. Depressurizing the saturated secondary side of the SGs results in a temperature reduction, which in turn reduces RCS temperature and pressure. The reduction in RCS temperature will result in inventory contraction in the RCS, with the result that the pressurizer would drain and a steam void would form in the reactor vessel upper head. The RCS leakage, particularly from the reactor coolant pump (RCP) seals, would also contribute to the decrease in RCS liquid volume. However, during the cooldown RCS pressure should drop below the safety injection accumulator pressure of about 630 psig, and the injection of some quantity of borated water into the RCS from the accumulators would then occur.

As discussed in its revised cooldown timeline, starting by approximately 28 hours into the event the licensee expects to further depressurize the SGs in several additional stages in order to further reduce RCS temperature and pressure. These additional SG depressurizations are expected to conclude by approximately 54 hours into the event, at which time the licensee expects to have fully opened the SG PORVs. In addition, as noted in the FIP, by approximately 6 days into the event, the licensee expects to use FLEX equipment from offsite response centers to restore the residual heat removal (RHR) system and supporting equipment, the operation of which would allow RCS temperature to be reduced below 200 degrees Fahrenheit (°F). Prior to undertaking the additional cooling and depressurization of the RCS, operators would need to perform a number of supporting actions including injecting additional boric acid into the RCS to avoid the potential for recriticality and isolating the accumulators using electrical power from FLEX generators to avoid the potential for excessive accumulator injection to the point that the nitrogen cover gas could enter the RCS.

Operators will ultimately transition the SG water supply from the TDAFW pump to portable FLEX pumps using water from the UHS, the standby nuclear service water pond (SNSWP).

The operators will perform dc bus load stripping within 3 hours following event initiation to ensure safety-related battery life is extended up to 18 hours. Following dc load stripping and

prior to battery depletion, one 500-kilowatt (kW), 600-volt alternating current (Vac) generator will be deployed from a FLEX storage building (FSB) to each unit. These portable generators will be used to repower essential battery chargers within 18 hours of ELAP initiation, as well as repowering containment hydrogen igniters, safety injection accumulator isolation valves, and portable FLEX sump pumps.

RCS makeup and boration will be initiated within 13 hours of the ELAP with loss of normal access to the ultimate heat sink event. Operators will provide reactor coolant makeup using portable FLEX high-pressure diesel-powered pumps, one per unit, to deliver water drawn from a FLEX connection on each refueling water storage tank (FWST) supply line. There is one FWST per unit. Borated water from the FWST will be injected into the RCS through FLEX connections to the safety injection system (one connection on either train). In addition, hoses can be routed from the pump discharge to the residual heat removal system FLEX connection.

If necessary, the volume of the FWSTs can be replenished in several ways including, (1) borated makeup from the concentrated solution in the boric acid tanks either directly or in a more dilute mixture with an unborated water source (e.g., raw water, National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) water purification equipment, an auxiliary feedwater condensate storage tank (CACST) (if available)), (2) the NSRC-supplied mobile boration skid, (3) the opposite unit's FWST, (4) the recycle holdup tanks (if available), and (5) portable FLEX drop tanks. The preferred option is to use water from existing tanks from the affected unit. The mobile boration skid enables mixing of powdered boric acid (also delivered from an NSRC) with water.

Each unit has an SFP, located in the unit's fuel building. Upon initiation of the ELAP event, the SFP will heat up due to the unavailability of the normal cooling system. The licensee has calculated that boiling could start as soon as 8.2 hours after the start of the event. To maintain SFP cooling capabilities, the licensee determined that it would take more than 24 hours for pool water level to drop to a level requiring cooling or the addition of makeup to preclude fuel damage. If necessary in Phase 1, makeup water can be provided via gravity drain from the FWST. To compensate for SFP boil-off in Phase 2, makeup water will be provided by pumping raw water from the SNSWP directly to the SFP operating floor through hoses. Makeup water can also be provided via hoses from fire protection piping or directly from fire protection piping if it is pressurized. This approach, which also requires SFP operating floor access, is simpler than running a hose from the SNSWP, but it is not credited since the fire protection piping is not hardened for all applicable external hazards. To vent boil-off steam from the SFP area, the FLEX response strategy directs the opening of the exterior roll-up door in each fuel building early in the ELAP event.

For Phases 1 and 2, the licensee's calculations demonstrate that no actions are required to maintain containment pressure below design limits for the initial 48 hours. In Phase 2, the operators will repower the hydrogen igniters inside containment to preclude the potential for hydrogen deflagration or detonation in the event of core damage. The igniters will be powered by the FLEX DGs. During Phase 3, the operators will provide long-term containment cooling by repowering a lower containment ventilation system fan within 48 hours of ELAP initiation to ventilate the hotter air within the SG and pressurizer enclosures. Within 52 hours, operators will also repower a containment air return fan to mix the colder air in the ice condenser with the rest of containment. These components will be powered using the large 4160 Vac diesel generator equipment provided by the NSRC.

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

3.2 Reactor Core Cooling Strategies

Order EA-12-049 requires licensees to maintain or restore cooling to the reactor core in the event of an ELAP concurrent with a loss of normal access to the ultimate heat sink. Although the ELAP results in an immediate trip of the reactor, sufficient core cooling must be provided to account for fission product decay and other sources of residual heat. Consistent with endorsed guidance from NEI 12-06, Phase 1 of the licensee's core cooling strategy credits installed equipment (other than that presumed lost to the ELAP with loss of normal access to the ultimate heat sink) 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 with loss of normal access to the ultimate heat sink 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 with loss of normal access to the ultimate heat sink 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 with loss of normal access to the ultimate heat sink 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 Duke's FIP [Reference 18], the heat sink for core cooling in Phase 1 would be provided by the four SGs at each unit, which would all be fed simultaneously by the unit's TDAFW pump with inventory supplied by either the unit's auxiliary feedwater storage tank

(CAST) or the underground raw water inventory remaining in the buried condenser circulating water system piping. The CAST is preferred for this purpose due to the higher quality of its water, but it is not protected against all applicable hazards; if it is available, it will have sufficient volume (about 300,000 gallons) to supply the TDAFW pump for approximately 18 hours. The underground circulating water piping will be automatically aligned to the TDAFW pump suction only if the CAST is unavailable, and its protected volume can provide enough SG feedwater for over 48 hours.

Duke's Phase 1 strategy directs the initiation of a cooldown and depressurization of the RCS within two hours of the initiation of the ELAP with loss of normal access to the ultimate heat sink event. During the audit, the licensee stated that simulator validation indicates that the actions necessary to begin the cooldown could be completed about 70 minutes after event initiation. Over a period of approximately 2 hours, Duke would gradually cool down the RCS from post-trip conditions until a SG pressure of 290 psig is reached, which results in an RCS cold-leg temperature of approximately 420 °F. Cooldown and depressurization of the RCS significantly extends the expected coping time under ELAP with loss of normal access to the ultimate heat sink 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. By terminating the initial RCS cooldown at a SG pressure of 290 psig, Duke has determined that injection of the nitrogen cover gas in the accumulators into the RCS will be prevented. The NRC staff's audit found the licensee calculational methods used in this determination to be appropriate.

3.2.1.1.2 Phase 2

Duke'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 will transition from supplying SG secondary inventory with the TDAFW pump to a portable, diesel-powered FLEX pump. This transition 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. To provide an unlimited source of secondary makeup in Phase 2, Duke stated that a low-pressure (LP) FLEX pump would be deployed at the credited water source and ultimate heat sink, the SNSWP. The LP FLEX pump will pressurize a raw water distribution header, from which a portable, diesel-driven, medium-pressure FLEX pump would take suction, supplying the SGs at a rate of up to 300 gallons per minute (gpm) at a SG pressure of 300 psig.

In addition to the primary core cooling strategy discussed above, Duke indicated that a number of alternate water sources exist. In particular, the CASTs and CACSTs have condensate-grade water which will not foul the SGs; however, these tanks are not protected from all applicable external hazards and have not been credited. Lake Norman may be available as a water source, but it is potentially vulnerable to a dam failure. The licensee's FIP states that raw water from Lake Norman or the SNSWP is acceptable for use as SG feed water for a limited duration of about 12 days. The licensee's strategy is to be on RHR cooling by 6 days after event initiation, after which SGs would not be needed for decay heat removal. If necessary, water purification equipment from the NSRC could be placed into service when it becomes available in Phase 3.

The FIP also states that portable FLEX diesel generators (DGs), power distribution panels, and cables will be placed into service in Phase 2 to provide electrical power to equipment needed to support the core cooling strategy, including the accumulator isolation valves, vital battery chargers, and portable FLEX sump pumps. Once FLEX power is established, 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, Duke'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. The licensee has submitted a revised estimate of the RCS cooldown profile [Reference 54] that is significantly more aggressive than the cooldown profile originally documented in the FIP. After cold leg accumulators have been isolated (calculated by the licensee to be completed no later than 28 hours into the event) operators will perform a second RCS cooldown to lower SG pressure to 160 psig, which corresponds to an RCS cold leg temperature of approximately 371 °F. This second cooldown, from approximately 420 °F to 371 °F in the RCS cold leg, should take less than an hour given the cooldown limit of 100 °F/hr. Therefore, the licensee states that the second cooldown will be completed by 29 hours into the event.

By 41 hours into the event, the licensee's revised cooldown strategy states that operators will further lower SG pressure to 80 to 100 psig, once a pressurizer PORV has been set up for long-term overpressure protection and SG feed has been transferred from the TDAFW pump to a diesel-driven FLEX pump (among other pre-conditions). By 54 hours into the event, operators plan to fully open the SG PORVs and cool the plant down as far as possible by releasing steam to the atmosphere.

Duke's FIP states that McGuire will 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. The FIP and its attached Schedule of Events timeline indicate that the transition to RHR cooling will commence no later than three days into the event and will be completed by no later than six days into the event. Transitioning within this timeframe allows management of excessive containment conditions (temperature, pressure, and sump level). The NSRC will provide a larger capacity low-pressure diesel-powered pump (5000 gpm at 150 psig discharge pressure), which will take suction from the SNSWP and be connected to the service water (SW) system to deliver cooling water to the component cooling water (CCW) heat exchanger. Operators will establish decay heat removal by first starting one train of CCW pumps along with the associated motor coolers. Once CCW flow is established, operators will align the RHR system and start one of its pumps to continue core cooling indefinitely.

McGuire will receive water purification equipment and a mobile boration skid from the NSRCs to ensure a long-term source of purified water. McGuire 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-MW 4160 Vac DGs per unit, which will allow repowering a 4160 Vac essential bus and required load centers. This increase in FLEX electrical power capacity will enable the repowering of specific installed plant equipment.

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 2 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 McGuire does not have a fully robust capability for 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, Duke 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 subcriticality throughout Phase 1, considering the planned cooldown profile.

Although not credited for any scenarios per the mitigating strategy documented in its FIP, the NRC staff's audit determined that McGuire has existing, installed equipment associated with its standby shutdown facility that the licensee considers robust in some ELAP scenarios and which would be the procedurally preferred source of RCS makeup in all ELAP scenarios if available. Specifically, within 10 minutes of event initiation, operators would attempt to use the diesel generator located in the standby shutdown facility to power a positive displacement pump having a capacity of approximately 26 gpm, which would supply inventory from the spent fuel pool to the RCS via the RCP seal injection lines. In scenarios where the standby shutdown facility is available, this uncredited means of providing RCS makeup is desirable for the duration that the condition of the spent fuel pool supports adequate pump suction because it would (1) maintain cooling to the RCP seals and (2) maintain flow across the RCP seal boundary into the RCS as opposed to allowing outward leakage of RCS inventory. Regardless of the licensee's not having formally credited this approach in its strategy for maintaining adequate RCS inventory, the NRC staff recognized the potential for this preferred means of RCS makeup to affect the spent fuel pool cooling strategy for an ELAP event. This issue is addressed subsequently in Section 3.3.4.3 of this safety evaluation (SE).

3.2.1.2.2 Phase 2

In order to maintain sufficient borated RCS inventory in Phase 2, Duke states that a diesel-driven, centrifugal, high-pressure FLEX pump would be deployed to inject borated coolant from the refueling water storage tank (FWST) into the RCS. The licensee will commence RCS makeup and boration no later than 13 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 the safety injection system FLEX connections; alternately, hoses can be routed from the FLEX pump discharge to FLEX connections on the RHR system.

The FWSTs (one at each unit) are robust to the applicable hazards identified in NEI 12-06, except for the portion extending above the 14-foot-high protective wall, which is vulnerable to wind-borne missiles. Each FWST, when intact, contains approximately 380,000 gallons of

borated water; if an FWST is struck by a wind-borne missile above the protective wall, the licensee stated that at least 116,000 gallons of usable inventory should still be available. The fully robust, creditable inventory in this tank should be sufficient to supply the high pressure FLEX pump for at least 1 to 2 days, depending upon the rate of injection needed to maintain RCS inventory. In the case of a wind-borne missile striking the exposed portion of the FWST, the preferred method of replenishing the FWST would be to pump spilled water from the annulus region (between the FWST and the protective wall) back into the tank using a 100-gpm sump pump. This pump would be powered by a portable FLEX 240 Vac generator deployed from one of the FSBs. The licensee determined that the annulus region may contain up to 80,000 gallons of usable water. For long-term makeup, or if the trapped water in the annulus region is not usable, Enclosure 6 of FSG-23, "Long Term FLEX Strategies," contains procedures for refilling the FWST from the following borated water sources:

- boric acid tanks (BATs)
- opposite unit's FWST
- portable FLEX boration mixing drop tanks and powdered boric acid
- recycle holdup tank (RHT)

Of these, only the BATs and the portable FLEX tanks are protected against all applicable hazards. The piping ties between the BATs and the FWSTs are not robust to the design-basis seismic hazard; however, in a seismic event the full FWST volume can be credited so makeup from the BAT in Phase 2 would not be required. The cross-connection between the two units' FWSTs is also not robust to the design-basis seismic hazard. The two BATs on site contain a combined usable volume of 70,000 gallons of highly borated water; procedure FSG-23 directs operators to blend the BAT inventory with non-borated water from either the FLEX raw water distribution header (which draws from the SNSWP and is protected) or the CACST (which is not protected). Use of the 3000-gallon portable FLEX tanks would entail mixing powdered boric acid, supplied by an NSRC, with one of these non-borated water sources.

The licensee indicated that the procedural target boron concentration for the blended borated coolant would be based on the objective of preventing recriticality under xenon-free Mode 5 conditions. The specific boron concentration to be used, which the licensee estimated to be in the range of 1200-1800 ppm, would be determined in the event by the licensee's reactor engineering group. Inasmuch as the blended boric acid concentration derived from the BAT inventory will not be sampled, the licensee allowed a 10 percent margin for flowrate uncertainty. The licensee further estimated that the solution in the BATs could provide a volume of blended coolant at the above concentrations in the range of 139,000 to 187,000 gallons per unit.

The NRC staff further noted that the licensee's strategy for replenishing the FWSTs with BAT inventory would involve the flow of highly concentrated BAT solution through a limited extent of outdoor piping and hoses in the vicinity of the FWST. Although the licensee correctly noted that wind-borne missile strikes are typically correlated with the warmer seasons, the staff nevertheless observed that, due to the high concentration of the BAT solution (i.e., which could precipitate in the range of 60-65 °F) and normal diurnal temperature variation, the potential for boric acid precipitation could not be ruled out. While the licensee noted that the applicable procedure would generally direct operators to maintain continuous flow from the BATs (once initiated), the staff's audit determined that the licensee's existing procedure does not adequately sensitize operators to the potential for boric acid precipitation in the flowpath from the BAT at

outdoor temperatures below the precipitation threshold for the BAT solution. The NRC staff determined that the NSRC-supplied mobile boration skid should be available by the time use of the BAT volume could become necessary, therefore the NRC staff concludes that this issue should not impact the overall success of the licensee's strategy to mitigate the analyzed ELAP event.

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. Operators can continue to replenish the volume of the FWST as described in Section 3.2.1.2.2. In addition, in Phase 3 the FWST can also be refilled using an NSRC-supplied mobile boration skid, or with raw water from the SNSWP filtered through an NSRC water purification skid and blended with powdered boric acid 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

The licensee's FIP states that Seismic Category 1 structures at MNS are not vulnerable to external flooding from the Probable Maximum Precipitation (PMP) or Probable Maximum Flood (PMF) event from the current licensing basis. Therefore, no variations to the core cooling strategy are necessary at this time. Refer to Section 3.5 below for the treatment of reevaluated hazards.

3.2.3 Staff Evaluations

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

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

3.2.3.1.1 Plant SSCs

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 licensee stated that safety-related or seismically rugged piping from the following systems are credited as part of the FLEX strategy and protected from the applicable external events: Reactor Coolant (NC) system, Auxiliary Feedwater (CA) system, Safety Injection (NI) system, Residual Heat Removal (ND) system, the Nuclear Service Water (RN) System, the Component Cooling (KC) system, the Main Steam (SM) System, the Instrument Air (VI) System, and the Condenser Circulating Water (RC) System. The licensee also stated that portions of the RC system would be modified so that the piping and components would be able to support FLEX strategies. 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 the applicable external hazards. There is one Auxiliary Building which is shared by both units. The TDAFW pump flow control valves (FCVs), which control flow to the SGs, are also located within the Auxiliary Building. The SG PORVs are used to remove heat during SG cooling, and they are located inside the Interior and Exterior Main Steam Doghouses, which are Seismic Category I structures. The FCVs and SG PORVs can also be operated with air from the instrument air blackout header and will remain available through air supplied by the FLEX air tanks during an ELAP until a diesel-powered FLEX air compressor is connected within 16 hours of event initiation.

The licensee described three water sources that are credited in its FIP as part of the FLEX strategy after ELAP. The CAST is described as the primary preferred source of SG feedwater because of its water quality, but it is not protected from all applicable hazards. The CAST contains approximately 300,000 gallons of demineralized water, which will provide approximately 18 hours of cooling water for the TDAFW pumps. The CAST may be refilled with raw water from other site sources or with treated water from NSRC water purification equipment during Phase 3. The licensee described the RC system pipe headers as the protected water source for SG makeup and is capable of providing 2 days of SG feedwater. The long-term SG makeup water source is described in the FIP as the SNSWP, which holds over 188 million gallons of water and allows for indefinite coping in light of the planned cooldown. The SNSWP is fully protected from the applicable external hazards and can also be used as makeup for the RCS and the SFP as needed during Phase 2 and Phase 3.

Based on the design and locations of the TDAFW pump, FCVs, and PORVs, the diverse water sources, as described above, and the SNSWP as described in the FIP, 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 AFW connection points, as described below in Section 3.7.3.1 and in the FIP, the NRC staff concludes that it appears that at least one of the connection points should be available to support core cooling using 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. There is one FWST per unit. The credited minimum inventory of the FWSTs is 383,146 gallons as required by Technical Specifications, with a minimum boron concentration of 2,675 ppm. The FWST is robust to all applicable hazards at MNS, except that the portion of the FWST above the 14 foot high protective wall around the tank is susceptible to high wind or tornado missiles. The licensee indicated that any damage to the above portion of the FWST would cause the water to spill into the FWST enclosure and can be pumped back into the protected portion of the FWST by a FLEX-deployed portable pump as needed. The licensee also stated that the FWST inventory can provide sufficient makeup capacity well into Phase 3. As described above in SE Section 3.2.1.2.2, the BAT is also available as a protected borated water source for high wind or tornado missile events, and will be blended with water from the SNSWP or from a CACST if it is available for RCS makeup.

Based on the design and availability of the FWST and BAT, as described in the FIP, 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, the NRC staff concludes that at least one of the connection points appears to be available to support RCS injection using a 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 Plant Instrumentation

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

- feedwater flowrate to SGs
- SG water level (narrow range)
- SG pressure
- RCS hot-leg temperature (wide range)
- RCS pressure (wide range)
- core exit thermocouples
- pressurizer level
- Reactor Vessel Level Indicating System
- wide-range neutron flux or source-range nuclear instruments

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.

The primary monitoring strategy for all of these parameters is to obtain readings from the main control room. 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 FSG-07, "Loss of Vital Instrumentation or Control Power."

3.2.3.2 Thermal-Hydraulic Analyses

Duke concluded that its mitigating strategy for reactor core cooling would be adequate based in part on a generic thermal-hydraulic analysis performed for a reference Westinghouse four-loop reactor using the NOTRUMP computer code. The NOTRUMP code and corresponding evaluation model were originally submitted in the early 1980s as a method for performing licensing-basis safety analyses of small-break loss-of-coolant accidents (LOCAs) for Westinghouse PWRs. Although NOTRUMP has been approved for performing small-break LOCA analysis under the conservative Appendix K paradigm and constitutes the current evaluation model of record for many operating PWRs, the NRC staff had not previously examined its technical adequacy for performing best-estimate simulations of the ELAP event. Therefore, in support of mitigating strategy reviews to assess compliance with Order EA-12-049, the NRC staff evaluated licensees' thermal-hydraulic analyses, including a limited review of the significant assumptions and modeling capabilities of NOTRUMP and other thermal-hydraulic codes used for these analyses. The NRC staff's review included performing confirmatory analyses with the 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 McGuire), 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, Revision 0, "Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances," industry has proposed defining this coping time as the point at which the one-hour centered time-average of the flow quality passing over the SG tubes' U-bend exceeds one-tenth (0.1). As discussed further in Section 3.2.3.4 of this evaluation, a second metric for ensuring adequate coping time is associated with maintaining sufficient natural circulation flow in the RCS to support adequate mixing of boric acid.

With specific regard to NOTRUMP, preliminary results from the NRC staff's independent confirmatory analysis performed with the TRACE code indicated that the coping time for Westinghouse PWRs under ELAP conditions could be shorter than predicted in WCAP-17601-P, "Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs." 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 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, McGuire 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 McGuire 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 McGuire 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. Consequently, the staff's audit considered the generic analysis from Sections 5.2.1 and 5.2.2 of WCAP-17601-P applicable to McGuire. 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. The licensee's strategy to provide RCS makeup by 13 hours into the event conservatively precedes the applicable time to reflux calculated in PWROG-14027-P by over 2 hours. However, the NRC staff noted that the time to reflux determined in PWROG-14027-P did not consider the potential for increased leakage due to seal faceplate degradation that could occur in an ELAP event due to the loss-of-seal cooling. This phenomenon is discussed in greater detail in the subsequent section of this evaluation. The NRC staff performed confirmatory calculations to determine the expected impact. On the basis of these calculations, the NRC staff concluded that, although the available margin would be reduced, providing RCS makeup by 13 hours into the event would still be capable of maintaining adequate natural circulation flow in the RCS.

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 McGuire unit use standard three-stage Westinghouse seal packages. As noted in Section 3.2.3.2, Duke is relying on thermal-hydraulic analysis

performed with the NOTRUMP code, as documented in WCAP-17601-P 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 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, Duke indicated that McGuire 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, "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," classify McGuire in the third generic analysis category (i.e., Category 3) 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.

The generic Westinghouse RCP calculations described above demonstrated that a Westinghouse PWR with a Category 3 leakoff line configuration would be expected to experience seal leakage significantly in excess of the predictions of WCAP-10541-P. Because the leakage rates from WCAP-10541-P were used as an input to the analyses in WCAP-17601-P, the licensee recognized that these analyses would not bound a Category 3 plant. Therefore, the licensee installed an additional 0.254-inch bore orifice in the first-stage seal leakoff lines for all four RCPs at each unit. These additional flow orifices are in series with the original 0.359-inch orifices, and are intended to limit seal leakage in an ELAP event and protect downstream piping and components from an associated pressure transient. The NRC staff performed confirmatory calculations to determine the expected impact. Accordingly, for the purpose of determining RCP seal leakage during the ELAP event, the NRC staff considers McGuire as having a Category 1 leakoff line configuration.

To ensure that the generic Category 1 leakage rates are applicable to McGuire, the NRC staff requested during the audit that Duke confirm that applicable portions of the first-stage seal leakoff line piping can withstand the maximum pressure experienced 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 for Duke 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 McGuire. During the audit, Duke informed the NRC staff that the applicable portions of the leakoff line piping and components can tolerate pressures greater than or equal to RCS design pressure (i.e., 2500 psia) at a fluid temperature of 568 °F. 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 McGuire.

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 mitigating strategy documented in McGuire's FIP does not satisfy Westinghouse's TB 15-1 recommendation. The licensee later submitted a revised estimate of the RCS cooldown profile [Reference 54] that is significantly more aggressive than the cooldown profile originally documented in the FIP. This cooldown profile more closely approaches, but still does not satisfy, the existing TB 15-1 recommendation. However, the PWROG is currently in the process of revising the recommended cooldown limits in TB 15-1 to address concerns with potential unintended consequences on TDAFW pump operation. During the audit, the licensee stated that it intends to implement the recommendations of the revised TB 15-1. The NRC staff intends to review the revised recommendations in the revised TB 15-1, when available, to ensure their adequacy.

The seal leakoff analysis discussed above assumes no failure of the seal design, including the elastomeric o-rings. During the audit review, Duke confirmed that all installed RCP seal o-rings at McGuire 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 McGuire under analyzed ELAP event conditions would not be expected.

During the audit review, Duke 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 indicates 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 indicates that the corrosion rate is temperature dependent. The NRC staff understands that the PWROG is currently working to address the potential for this

phenomenon to result in gradually increasing leakage rates from Westinghouse-style RCP seals.

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 13 hours would ensure that natural circulation can be maintained in the RCS. McGuire's initial RCS makeup flow capacity of 90 gpm significantly exceeds the total rate of RCS leakage that the PWROG's analysis predicts following RCS depressurization, such that RCS inventory would begin to recover upon restoration of RCS makeup. At the time of developing the mitigating strategy documented in its FIP, however, the licensee was not fully aware of the potential for hydrothermal corrosion to increase the long-term seal leakage rate.

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 beyond the available FLEX injection capacity. According to McGuire's FIP, plant operators would cool the RCS to 350 °F in the RCS cold legs within approximately 48 hours, as opposed to the 24 hours currently recommended by Westinghouse in TB 15-1. 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. Following discussion of these concerns during the audit, the licensee submitted a revised estimate of the RCS cooldown profile realistically achievable during the analyzed ELAP event [Reference 54] that was significantly more aggressive than the cooldown profile documented in McGuire's FIP. In particular, the revised cooldown profile included additional stages of cooldown and depressurization, including fully opening the SG PORVs by 54 hours into the event. Completing the additional stages of RCS cooldown and depressurization should have several beneficial effects, including halting the hydrothermal corrosion reaction and significantly reducing RCS leakage. Although the licensee's strategy does not credit reclosure of the seal leakoff line relief valve, fully opening the SG PORVs as directed by the revised cooldown profile may reduce RCS pressure below the 150-psig nominal set pressure of this relief valve, which could allow the relief valve to reseat and essentially terminate leakage from the RCP seals. Regardless of whether the relief valve reseats, the licensee would obtain the NSRC-supplied mobile boration skid, which should provide sufficient capacity to supply borated coolant for RCS makeup for an indefinite period once the SGs have been fully depressurized. According to the revised RCS cooldown profile, the NRC staff performed additional confirmatory RCS mass balance calculations estimating the expected seal leakage rate during the analyzed ELAP event as a function of time relative to the available FLEX RCS injection capacity. The NRC staff's confirmatory calculations showed that, for the analyzed ELAP event, (1) the more aggressive cooldown profile should terminate the hydrothermal corrosion reaction in sufficient time to maintain the RCS leakage rate within the available FLEX RCS injection capacity, and (2)

available quantities of borated coolant for FLEX RCS injection should be sufficient to maintain adequate RCS inventory.

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

3.2.3.4 Shutdown Margin Analyses

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

- the cooldown of the RCS and fuel rods adds positive reactivity
- the concentration of xenon-135, 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.

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 notes that any core design changes, such as those considered in a core reload analysis, should be evaluated to determine that they do not adversely impact the approved FLEX strategies, especially the analyses which demonstrate that recriticality 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 13 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 recriticality

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 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. According to this method, using the RETRAN-3D Mod 4.7 computer code as recorded in Appendix D of calculation DPC-1552.08-00-0278, Revision 3, the licensee calculated that less than 9.7 hours would be necessary to borate the RCS to a concentration that would ensure adequate shutdown margin at 350 °F (i.e., boration completed by 22.7 hours into the ELAP event). The calculation in Appendix D incorporated a concurrent RCS cooldown beginning at 22 hours into the event, which is not part of Duke's strategy. Considering the licensee's current mitigating strategy, which does not direct a concurrent cooldown, an additional letdown cycle using the reactor vessel head vent flowpath would be necessary to inject the required quantity of boron, thereby delaying completion of this activity to 24 hours into the event. Duke further calculated that 14,970 gallons of FWST water, which is borated to a minimum concentration of 2675 ppm, is required to ensure long-term subcriticality 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 116,000-gallon approximate usable volume that the licensee calculated would remain in the FWST even assuming the worst-case missile damage.

Toward the end of an operating cycle, when the initial 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 licensee discussed McGuire's capability to conduct RCS venting in the case that letdown from the RCS is necessary. The licensee stated that, in this case, operators would follow the direction of guideline FSG-8 to open the reactor vessel upper head vent path, which contains dc-powered valves that can be operated from the control room under ELAP conditions. The licensee indicated that 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 assumed required final boron concentration in the RCS is 475 ppm, which Duke stated contains margin to account for expected variations in future core designs.

- 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 staff noted that the available RCS makeup flowrate during this phase of the event would be 90 gpm, as opposed to the 40 gpm conservatively assumed by the licensee. 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, the actual pressure would initially be expected to exceed this value; and further, in a case corresponding to zero RCS leakage, refilling the vessel head and restoring pressurizer level should result in RCS pressurization and subcooling 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 hours into the ELAP event.

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 (ADAMS Accession No. ML13276A183), the NRC staff endorsed the above position paper with three conditions:

- The required timing and quantity of borated makeup should consider conditions with no RCS leakage and with the highest applicable leakage rate.

- Adequate borated makeup should be provided either (1) prior to the RCS natural circulation flow decreasing below the flow rate corresponding to single-phase natural circulation, or (2) if provided later, then the negative reactivity from the injected boric acid should not be credited until one hour after the flow rate in the RCS has been restored and maintained above the flow rate corresponding to single-phase natural circulation.

- A delay period adequate to allow the injected boric acid solution to mix with the RCS inventory should be accounted for when determining the required timing for borated makeup. Provided that the flow in all loops is greater than or equal to

the corresponding single-phase natural circulation flow rate, a mixing delay period of one hour is considered appropriate.

During the audit review, Duke confirmed that McGuire will comply with the August 15, 2013, position paper on boric acid mixing, including the conditions imposed in the staff's corresponding endorsement letter.

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 its FIP the use of a FLEX low-pressure pump (diesel-powered booster pumps) located at the SNSWP with the function of supplying water from the SNSWP to the suction of a FLEX medium-pressure pump located in the station yard. The pump hoses connected to the FLEX low-pressure pump can be routed through either the North or South vehicle access portals. The FLEX medium-pressure pump (diesel-powered) is supplied by the FLEX low-pressure pump and can provide water from the CAST or the SNSWP to the SGs. One low-pressure pump and one medium-pressure pump is needed for each unit. The licensee indicated that MNS has three portable FLEX low-pressure pumps and three portable FLEX medium-pressure pumps, stored in the FLEX Buildings, to satisfy the N+1 inventory requirement. The deployment of the FLEX low and medium-pressure pumps will begin when manpower becomes available during the ELAP event, so that SG makeup will be available from the SNSWP if needed. If the TDAFW pump fails prior to staging these pumps, the FLEX pump deployment will be prioritized as directed by procedure ECA-0.0. For RCS makeup, the licensee described in the FIP the use of a FLEX high-pressure pump (diesel-powered, centrifugal pump) to provide RCS boration from the FWST. The licensee also stated that MNS has three portable FLEX high-pressure pumps, stored in the FLEX Buildings, to satisfy the N+1 inventory requirement. The deployment of the FLEX high-pressure pumps will be established 9 hours after ELAP is declared so that the FWST can be used to provide RCS makeup for Phase 2. The licensee also described FLEX sump pumps to mitigate potential internal flooding of areas in the auxiliary building containing equipment needed for the FLEX response strategies after 18 hours. The licensee stated that MNS has six submersible FLEX sump pumps, with each of the three FLEX storage buildings containing two pumps. One of the sump pumps is diesel-powered. The other five pumps are electrically powered and require that FLEX power be established to enable use of the pumps. The licensee plans to establish the FLEX electrical distribution system within 12 hours. The water sources for all three pumps to be used for SG and RCS makeup are described with more detail in SE Section 3.10 below.

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 the following FLEX hydraulic calculations to indicate that the above portable FLEX pumps were capable of performing their functions after a BDBEE event:

- Calculation MCC-1240.00-00-0009, Rev. 0, "FLEX Mitigation Strategy – Westinghouse Evaluation DAR-SEE-11-13-12, Evaluation of Alternate Coolant Sources for Responding

to a Postulated Extended Loss of AC Power," which evaluated the use of the FLEX pumps using makeup water from the SNSWP to supply the SGs. The licensee has three FLEX low-pressure pumps, in which the #1 and #2 pumps can each supply a design flow of 1,500 gpm. The #3 pump can supply a design flow of 3,000 gpm (potentially supplying both units) when taking suction on the SNSWP. The FLEX medium-pressure pump is a 300 gpm, centrifugal pump with a maximum discharge pressure of 400 psig.

- Program document MCS 1465.00-00-0026, Rev.1, "Design Basis Specification for the Flexible Response to Extended Loss of All AC Power," which evaluated the use of FLEX pumps providing makeup to the RCS from the FWST or BAT.

The staff also conducted a walkdown 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 MNS, as described in the above hydraulic analyses and the FIP, the NRC staff concludes that the portable FLEX pumps should perform as intended to support core cooling and RCS inventory control during an ELAP event, consistent with NEI 12-06, Section 11.2.

3.2.3.6 Electrical Analyses

The licensee's FIP defines strategies capable of mitigating a simultaneous loss of all ac power and loss of normal access to the ultimate heat sink resulting from a BDBEE by providing the capability to maintain or restore core cooling (the licensee's strategy for RCS inventory control uses the same electrical strategy as for maintaining or restoring core cooling, containment, and SFP cooling) at both units on the McGuire site. Furthermore, the electrical coping strategies are the same for all modes of operation.

The MNS electrical FLEX strategies are practically identical for each unit for maintaining or restoring core cooling, containment, and spent fuel pool cooling, except as noted in Sections 3.3.4.4 and 3.4.4.4 of this SE.

The NRC staff reviewed the licensee's FIP to determine whether the FLEX strategies, if implemented appropriately, would maintain or restore core cooling, containment, and spent fuel pool cooling following a BDBEE. As part of its review, the NRC staff reviewed 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 credited in the FIP as a result of loss of heating, ventilation, and air conditioning (HVAC) during an ELAP as a result of a BDBEE.

According to the licensee's FIP, the MNS 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 with a simultaneous loss of access to the ultimate heat sink. In its FIP, the licensee assumes that this determination can be made within 2 hours after the onset of an ELAP.

During the first phase of the ELAP event, MNS 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 (reactor core cooling, RCS inventory control,

and containment integrity). The MNS 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 external hazards. Each of the two units at MNS are provided with four 125 Vdc safety-related batteries (EVCA, EVCB, EVCC, and EVCD). The safety-related batteries are arranged as A Train (EVCA, EVCC) and B Train (EVCB, EVCD). The safety-related batteries use GNB NCN-27 battery cells and each battery is rated for 1944 ampere-hours (AH) at an 8 hour discharge rate to 1.75 V per cell. To extend the coping capability of the vital station batteries, Duke will complete load shedding within 3 hours of ELAP initiation to reduce battery discharge to only essential loads (e.g., vital instrumentation). This action will extend the functional capability of the vital station batteries to at least 18 hours to allow connection of the FLEX portable DGs.

The licensee performed a dc coping study, MCC-1381.05-00-0351, "U1/2, 125 VDC Vital I&C Power System (EPL) Battery Coping SBO Coping Time Estimate, INPO IER L1-11-04," Rev. 3, which verified the capability of the dc system to supply the required loads during the first phase of the MNS FLEX mitigation strategy plan for an ELAP as a result of a BDBEE. The licensee proposed two 125 Vdc bus crosstie strategies to extend the vital battery coping period greater than 8 hours. These strategies include a dead bus battery transfer and a hot bus battery transfer.

The proposed strategy for the hot bus transfer is to load strip and crosstie EVCA to EVCC batteries, and EVCB to EVCD batteries. Closing a dc tie breaker between dc buses EVDA/EVDC and EVDB/EVDD could result in a significant electrical arc if there is a significant difference in voltage between batteries EVCA/EVCC and EVCB/EVCD at the time of the breaker closure. Based on this concern, the staff asked the licensee to discuss any adverse impacts as a result of closing the dc tie breakers during an ELAP event. The staff's review of the design basis calculations indicated that the battery terminal voltages track together throughout the battery cycle and that the terminal battery voltage difference is minimal. Based on the evaluation above, the dc cross-tie breakers should close as expected when implementing the cross-tie between dc buses EVDA/EVDC and EVDB/EVDD.

The proposed strategy for the dead bus battery transfer is to take out of service the EVCB and EVCC vital batteries at 2 hours into the event, and finish load stripping at 3 hours. The batteries will be disconnected from their distribution centers (EVDB and EVDC) in order to preserve battery life for later use. For the 'A' train safety-related batteries the calculation showed that opening the breaker to battery EVCC at t = 3 hours and finishing load stripping at t = 3 hours, operating 125 Vdc bus EVDA initially from the EVCA battery, and then connecting bus EVDA to battery EVCC and using the remaining charge on battery EVCC results in a total coping time of approximately 20.2 hours (10.5 hours using battery EVCA and 9.7 hours using battery EVCC). For the 'B' train safety-related batteries the calculation showed that opening the breaker to battery EVCB at t = 3 hours and finishing load stripping at t = 3 hours, operating 125 Vdc bus EVDD initially from the EVCD battery, and then connecting bus EVDD to battery EVCB results in a total coping time of approximately 21.5 hours (11.6 hours using battery EVCD and 9.96 hours using battery EVCB). Duke expects that power will be restored to the battery chargers within 18 hours. Duke stated that the NEI white paper, "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern," ADAMS Accession No. ML13241A186, which was endorsed by the NRC (see ADAMS Accession No. ML13241A188), is applicable to MNS.

In addition to the NEI white paper, the NRC sponsored testing at Brookhaven National Laboratory that resulted in the issuance of NUREG/CR-7188, "Testing to Evaluate Extended Battery Operation in Nuclear Power Plants," in May of 2015. The purpose of this testing was to examine whether existing vented lead acid batteries can function beyond their defined design-basis (or beyond-design-basis if existing Station Blackout (SBO) coping analyses were utilized) duty cycles in order to support core cooling. The study evaluated battery performance availability and capability to supply the necessary dc loads to support core cooling and instrumentation requirements for extended periods of time.

The testing provided an indication of the amount of time available (depending on the actual load profile) for batteries to continue to supply 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 manufacturer performance data. According to the NUREG, the projected availability of a battery can be accurately calculated using the Institute of Electrical and Electronics Engineers (IEEE) Standard 485-2010, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," or using an empirical algorithm described in the report.

Based on the evaluation above, the NRC staff concludes that the MNS load shed strategy should ensure that the batteries have sufficient capacity to supply power to required loads for at least 18 hours.

During Phase 2, the licensee will deploy, stage, and connect a FLEX 600 Vac DG for each unit to repower 600 Vac buses to ensure power is available to loads, such as the battery chargers prior to depletion of the station batteries, containment hydrogen igniters, cold leg accumulator isolation valves, and portable FLEX sump pumps. As identified previously, the licensee will perform a manual load shedding of the station batteries to ensure that the station batteries remain available until the FLEX 600 Vac DGs repower the 600 Vac buses. The FLEX 600 Vac DGs are 500 kilowatt (kW) standby rating generators that are trailer-mounted with a double-walled diesel fuel tank built into the trailer which can hold 12 hours full load fuel supply. Three FLEX 600 Vac diesel generators are available in order to satisfy the N+1 requirement.

The NRC staff reviewed the licensee's Phase 2 and Phase 3 FLEX generator calculations and procedures (MCC-1381.05-00-0352, "U1/U2 Alternate AC System Deployment for Extended Loss of AC Power (ELAP)", Rev 2, "Duke Fleet FLEX Diesel Generator Procurement Specifications" (DPS-1612.00-0001, CNS-1612.00-0001, MCS-1612.00-0001, OSS-0351.00-00-0013), Rev. 0, FG/0/A/FLEX/FSG-20, "Flex Electrical Distribution", Rev. 1, and FG/0/A/FLEX/FSG-04, "ELAP DC Bus Management", Rev. 1, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the Class 1E equipment. Based on the NRC staff's review, the minimum required loads for Phase 2 equate to 300 kW each for Unit 1 and Unit 2. Therefore, one 500 kW FLEX Diesel Generator per unit is adequate to support the electrical loads required for either the primary or the alternate 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 becoming fully discharged.

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 (two per unit) 1-megawatt (MW) 4160 Vac 3-phase turbine generators, two (one per unit) 1 MW 480 Vac 3-phase turbine generators, 480 Vac to 600 Vac transformers (one per unit), and 4160 Vac distribution panels (one per unit, including cables and connectors). The 4160 Vac turbine generators are capable of supplying approximately 1 MW each, but two turbine generators will be operated in parallel to provide approximately 2 MW for each unit. Based on the licensee's calculation, MCC-1381.05-00-0352, "U1/U2 Alternate AC System Deployment for Extended Loss of AC Power (ELAP)", Rev 2, the minimum required loads equate to 1.6 MW each for Unit 1 and Unit 2. Sufficient margin exists between the calculated loading and the capacity of the 4160 Vac turbine generators being supplied by the NSRC to ensure that the minimum required loads can function as expected. The capacity of the NSRC-supplied 480 Vac turbine generator is of greater capacity than the capacity of the licensee's Phase 2 FLEX DGs. Therefore, the NRC staff finds that the Phase 3 4160 Vac and 480 Vac turbine generators will have adequate capacity to supply power to the electrical loads used in Phase 2 to maintain or restore core cooling, SFP cooling, and containment indefinitely following an ELAP.

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

3.2.4 Conclusions

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

3.3 Spent Fuel Pool Cooling Strategies

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 spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load; and 3) spray via portable monitor nozzles from the refueling floor using a portable pump capable of providing a minimum of 200 gallons per minute (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 must be completed within a certain period of time should be identified and a basis that the time can be reasonably met should be provided. 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.

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

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

3.3.1 Phase 1

The licensee stated in its FIP, that no actions are required during ELAP Phase 1 for SFP makeup because the time to boil is sufficient to enable deployment of Phase 2 equipment. Adequate SFP inventory exists to provide personnel shielding well beyond the time of boiling. The licensee will monitor SFP water level using reliable SFP level instrumentation installed per Order EA-12-051. The licensee stated that it can provide SFP makeup through gravity drain from the FWST for Phase 1 if necessary.

3.3.2 Phase 2

The licensee stated in its FIP that SFP makeup will be provided from the SNSWP directly to the SFP deck through hoses. The hoses will be connected to a spray header (Boggs Box) about 18 hours after ELAP is declared when access to the SFP deck is less challenging. The licensee stated that the alternate strategy is to connect hoses from the SNSWP to the Spent Fuel Cooling (KF) system suction lines. This connection will allow makeup to the SFP without access to the SFP deck. Both connections are described in more detail in SE Section 3.7.3.1. Additional SFP makeup methods which may be available as described in the FIP are adding borated water from the FWST to the SFP using the existing piping and/or connecting hoses to fire protection piping or directly from fire protection piping if it is pressurized.

3.3.3 Phase 3

The licensee stated in its FIP that water purification equipment from the NSRC will be available to the site to ensure a long-term source of clean water for SFP cooling while utilizing the SNSWP.

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 in the FIP as the plant SSC needed for SFP cooling. The SFP Building is described as a mostly Seismic Category I structure that is designed to provide protection from the applicable external hazards. Portions of the north end of the SFP Building are not robust to all applicable hazards. The licensee evaluated these locations and determined that the lack of protection from all applicable hazards does not affect the SFP cooling strategy.

The staff reviewed the licensee's calculation on habitability on the SFP refuel floor. During the audit review, the licensee provided program document MCS 1465.00-00-0026, Rev.1, "Design Basis Specification for the Flexible Response to Extended Loss of All AC Power." The program document and the FIP indicate that boiling begins at approximately 35 hours during a normal, non-outage situation. The staff noted that the licensee's sequence of events timeline in its FIP indicates that operators will deploy hoses and spray nozzles as a contingency for SFP makeup around 18 hours after event initiation to ensure the SFP area remains habitable for personnel entry.

As described in the licensee's FIP, the licensee's Phase 1 SFP cooling strategy does not require any operator actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of the SFP. The operators are directed to open the SFP exterior roll-up door about 18 hours after ELAP is declared to establish a steam vent path from the SFP Building during an ELAP event. This action will minimize the impact of condensed steam from the SFP on SFP Building habitability. The operators will monitor radiation levels outside the SFP roll-up door. This action is captured in procedure FSG-11, Rev.1, "Initial Assessment and FLEX Equipment Staging."

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves the use of the FLEX low-pressure pump with associated hoses and fittings taking suction from the SNSWP and discharging to the SFP. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX pump is discussed in Section 3.7.3.1 below. Furthermore, the staff's evaluation of the robustness and availability of the SNSWP for an ELAP event is discussed in Section 3.10.3.

3.3.4.1.2 Plant Instrumentation

In its FIP, the licensee stated that the instrumentation for SFP level will meet the requirements of Order EA-12-051. Furthermore, the licensee stated that these instruments will have local battery power which is expected to last about 7 days. Guidance is provided in FSG-05 to change batteries or supply alternate power after the batteries deplete. The NRC staff's review of the SFP level instrumentation, including the primary and back-up channels, the display to

monitor the SFP water level and environmental qualifications to operate reliably for an extended period are discussed in Section 4 of this SE.

3.3.4.2 Thermal-Hydraulic Analyses

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 referenced calculation MCC-1201.28-00-0001, Rev. 1, "Evaluation of Potential Boron Dilution Accidents for the McGuire Spent Fuel Pools," and program document MCS 1465.00-00-0026, Rev.1, "Design Basis Specification for the Flexible Response to Extended Loss of All AC Power." The licensee evaluated the SFP with the maximum design-basis heat loads (i.e. one-third recently discharged core offload at 150 hours) and the maximum allowable initial pool temperature (140°F) to conclude that the minimum time to boiling is 8.2 hours. The licensee also determined that using a best estimate of approximately 20 days of heat loads and the maximum pool temperature, the minimum time is 17 hours. For normal conditions of initial pool temperature of 90 °F immediately following a typical 21-day refueling outage, the licensee concluded that the minimum time to boil is extended to about 35 hours. The licensee concluded in the above documents that the required SFP makeup overflow occurs after 15 hours, and the time to reach the minimum required boron concentration of 730 ppm is over 80 hours. The staff evaluated both documents 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 both SG makeup and SFP makeup as needed.

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 of the onsite FLEX pump if the onsite FLEX low-pressure pump were to fail. As stated in the FIP, the SFP makeup rate of 400 gpm meets or exceeds the maximum SFP makeup requirements.

The licensee also described during the audit an alternative method of providing cooling to the RCP seals within ten minutes of ELAP. The Standby Shutdown Facility DG may be able to be started. If so, it will provide power to the positive displacement standby makeup pump (SBMUP) located in the reactor building annulus, which takes a suction from the SFP using the

transfer canal, and provides seal injection flow to the RCP seals. The licensee indicated in its evaluation of the SFP that supplying the SBMUP for about 20 hours without SFP makeup would cause a reduction of the SFP inventory of about 2 feet (30,000 gallons). Guideline FSG-11, "Alternate SFP Make-up/Cooling Strategy," instructs operators to begin makeup to the SFP either using the FWST if available or the SNSWP around the 20 hour time frame after ELAP is declared. The Boggs Box used for SFP makeup is evaluated to be capable of providing for the loss of the SFP water within 1 hour after deployment. The licensee also requires operators to monitor SFP level in procedure ECA-0.0, in which if the SFP drops more than 1 foot below normal level, the operators are instructed to begin SFP makeup. This is conservative compared to the general criteria for an ELAP stated in NEI 12-06, Section 3.2.1.1, which is to keep the fuel in the SFP covered with water. Normally there is about 23 feet of water above the fuel. The staff finds that the utilization of the SFP for supplying the SBMUP is appropriate due to the monitoring of the SFP after ELAP is declared and the licensee's capability to deploy FLEX equipment and initiate SFP makeup from the FWST or SNSWP when SFP levels drop more than 1 foot below the normal level.

3.3.4.4 Electrical Analyses

The licensee's OIP and FIP define strategies capable of mitigating a simultaneous ELAP and loss of normal access to the ultimate heat sink resulting from a BDBEE by providing the capability to maintain or restore core cooling, containment, and SFP cooling at the McGuire site. The electrical coping strategies are the same for all modes of operation.

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

3.3.5 Conclusions

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

3.4 Containment Function Strategies

The industry guidance document, NEI 12-06, Table 3-2, 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 MNS 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", Rev. 2, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of repowering a Hydrogen Skimmer Fan 24 hours after an ELAP-inducing event, two Lower Containment Fans 48 hours following an ELAP-inducing event, and a Containment Air Return Fan 52 hours after the ELAP-inducing event. The strategy is described and proceduralized in FSG-12, "Alternate Containment Cooling", Rev. 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 1.1 psig for an 8 hour period in the 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 hydrogen igniters can be repowered within approximately 1 hour after the FLEX Electrical Distribution System is set up at 14 hours following an ELAP event. Actions to repower hydrogen igniters is controlled in guideline FSG-05, based on priority and status of core cooling.

3.4.1 Phase 1

The MNS 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.

3.4.2 Phase 2

The MNS 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 approximately 14 hours following the initiating event. The associated motor control center to energize the hydrogen igniters can be repowered within approximately one hour after the FLEX Electrical Distribution System is set up. Providing power to the containment hydrogen igniter assemblies would prevent the buildup of hydrogen gas, even though core damage is not expected during this ELAP.

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.

3.4.3 Phase 3

During Phase 3, the NSRC-supplied 4160 Vac generators will be available to provide power to the Lower Containment Ventilation fans to ventilate the hotter air 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.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

Guidance document NEI 12-06 baseline assumptions have been established on the presumption that other than the loss of the ac power sources and normal access to the UHS,

installed equipment that is designed to be robust with respect to design-basis external events is assumed to be fully available. Installed equipment that is not robust is assumed to be unavailable. Below are the baseline assumptions for the availability of SSCs for maintaining containment functions during an ELAP.

3.4.4.1.1 Plant SSCs

Sections 1.2.2.4 and 3.8.2.1 of the MNS 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 Class I structures. Table 3-1 further shows that the Reactor Building has been designed to protect against tornado loads and missiles.

The Lower Containment Ventilation Fans are non-safety-related equipment which the licensee stated are designed to Duke Class H (Quality Class 4). The NRC staff requested more information about the robustness of these fans, and the licensee provided calculation MCM-1223.42-0005, "Seismic Robustness Review of McGuire Unit 2 Non-Safety Condenser Circulating Water (RC) System for Assured Water FLEX Modification", Rev. 0, which contained an evaluation of the Lower Containment Ventilation Fans that demonstrated the equipment was seismically anchored and also stated that the fans are connected to Class 1E electrical power.

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 6.2.7 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.

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 would be available following an ELAP-inducing event. Additionally, the evaluation for the Lower Containment Ventilation Fans shows that they are robust, as defined by NEI 12-06, and would 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", Rev. 2, 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, Rev. 1. Section 3.2.3.3 of this SE contains more information about the RCP seals and leakage.

Case 1 of the calculation modeled the behavior of the containment parameters with no mitigation actions being taken. Under these conditions, the temperature in 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, two Lower Containment Fans at 48 hours, and 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 295 °F at hour 48. Repowering the two Lower Containment Fans 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 repowering of the two Lower Containment Fans 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 MNS 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 FLEX Pumps and Water Supplies

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 Electrical Analyses

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. The licensee's evaluation concluded that containment pressure and temperature remain at or below acceptable values during the initial 24 hours after the event. Therefore, actions to reduce containment temperature and pressure and to ensure continued functionality of the key parameters will not be required immediately.

The Phase 1 coping strategy for containment involves monitoring containment pressure using the Containment Wide Range Pressure Instrumentation and no other actions are required during Phase 1. The licensee also stated that the ice condenser containment design helps maintain containment conditions in all phases of the ELAP event, until the ice bed inventory is depleted.

The Phase 2 coping strategy is to use the FLEX Electrical Distribution System and the FLEX DGs to repower Hydrogen Skimmer fans within 24 hours of the start of the ELAP, which help limit the temperature increase in the SG and Pressurizer enclosures. 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. The licensee analyses shows that containment pressure and temperature will remain at or below acceptable values during the initial 48 hours after the event, but in order to mitigate adverse containment conditions, Phase 3 FLEX strategies are required to be implemented within 48 hours of ELAP initiation. The licensee will continue to monitor containment pressure using the Containment Wide Range Pressure Instrumentation.

The Phase 3 coping strategy is to use the 4160 Vac turbine diesel generators provided by the NSRC to repower a lower containment ventilation fan within 48 hours to ventilate the hotter air within the SG and Pressurizer enclosures and a containment air return fan within 52 hours to mix the colder air in the ice condenser with the rest of containment.

The NRC staff reviewed MCC-1381.05-00-0352, "U1/U2 Alternate AC System Deployment for Extended Loss of AC Power (ELAP)", Rev 2, conceptual single line electrical diagrams, the separation and isolation of the FLEX turbine generators from the Class 1E emergency diesel generators (EDGs), and procedures that direct operators how to align, connect, and protect

associated systems and components. Based on its review of the sizing calculation, the NRC staff confirmed that two 1 MW 4160 Vac turbine generators per unit will provide sufficient capacity and capability to supply the necessary loads following an ELAP to maintain core cooling and containment. Refer to Section 3.2.3.6 above for additional analysis.

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, Revision 0, 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 ultimate heat sink.

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 interim staff guidance in JLD-ISG-2012-01 [Reference 7]. Coincident with the issuance of the order, on March 12, 2012, the NRC staff issued a Request for information Pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) [Reference 19] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in responses to the requested information and the requirements for Order EA-12-049 and related rulemaking to address beyond-design-basis external events (see COMSECY-14-0037, Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards" [Reference 47]). The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 20]. The Commission approved the staff's recommendations that licensees need to address the reevaluated flooding hazards within their mitigating strategies for 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 35], the NRC staff informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the related NRC SEs and inspections will rely on the guidance provided in JLD-ISG-2012-01, Rev. 0 [Reference 7] and the related industry guidance in Revision 0 to NEI 12-06 [Reference 6]. The reevaluations may also identify issues to be entered into corrective action programs consistent with the OIPs submitted in accordance with Order EA-12-049.

The licensee has submitted its flood hazard reevaluation report (FHRR) [Reference 21], and the NRC staff has completed a review and the results are discussed in Section 3.5.2 below. The licensee developed its OIP for mitigation strategies in February 2013 [Reference 10] by considering the guidance in NEI 12-06 and its then-current design-basis hazards. Therefore, this SE makes a determination based on the OIP and FIP, and notes the possibility of future actions by the licensee if the licensee's FHRR identifies a flooding hazard that exceeds the current design-basis flooding hazard.

Per the 50.54(f) letter, licensees were also asked to provide a seismic hazard screening and evaluation report (SHSR) to reevaluate the seismic hazard at their site. The licensee submitted its SHSR [Reference 22] in March 2014, and the NRC staff has completed a review and the results are discussed in Section 3.5.1 below. Therefore, this SE makes a determination based on the OIP and FIP, and notes the possibility of future actions by the licensee if the licensee's SHSR identifies a seismic hazard which exceeds the current design-basis seismic hazard.

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

3.5.1 Seismic

In its FIP, the licensee stated that seismic hazards are applicable to the site. In its SHSR, the licensee stated that per UFSAR Section 3.1, the design-basis earthquake (DBE) seismic criteria for the site is fifteen-hundredths 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 a certain frequency range, such as the numbers above, is often used as a shortened way to describe the hazard. The NRC's current terminology for the DBE is the safe shutdown earthquake (SSE).

As previously discussed, the NRC issued a 50.54(f) letter [Reference 19] that requested facilities to reevaluate the site's seismic hazard. In addition, the 50.54(f) letter requested that licensees submit, along with the hazard evaluation, an interim evaluation and actions planned or taken to address the reevaluated hazard where it exceeds the current design-basis seismic hazard.

Based on the results of its SHSR, the plant screened-in for a risk evaluation, a high frequency evaluation, and a SFP evaluation. The high frequency evaluation will be done as part of the risk evaluation. Electric Power Research Institute (EPRI) Report 3002000704 [Reference 40], referred to as the Augmented Approach, was developed as the process for evaluating

selected critical plant equipment prior to completing plant seismic risk evaluations. The NRC endorsed this report by letter dated May 7, 2013 [Reference 41]. The Augmented Approach outlines a process for responding to the seismic evaluation requested in the 50.54(f) letter under Recommendation 2.1, "Seismic." The process includes a near-term expedited seismic evaluation process followed by plant risk evaluations in accordance with EPRI Report 1025287 [Reference 42]. This Augmented Approach ensures that installed Phase 1 plant equipment and Phase 2 and 3 plant connections credited for FLEX strategies would retain function during and after a beyond-design-basis seismic event using seismic margins assessment criteria, by calculating a High Confidence of Low Probability of Failure (HCLPF) seismic capacity and comparing that to the seismic demand of a Review Level Ground Motion (RLGM), capped to two times the SSE in the frequency range of 1 to 10 Hertz (Hz). This provides assurance of plant safety while the plant completes the seismic probabilistic risk assessment (SPRA).

The NRC staff performed a screening and prioritization determination of the MNS SHSR, as documented by letter dated May 9, 2014 [Reference 48].

The NRC staff completed its review of the licensee's SHSR, as documented by letter dated July 20, 2015 [Reference 43]. The staff concluded that the licensee conducted the hazard reevaluation using present-day methodologies and regulatory guidance, appropriately characterized the site given the information available, and met the intent of the guidance for determining the reevaluated seismic hazard. The staff also concluded that the reevaluated seismic hazard for the plant is suitable for other activities associated with the NTTF Recommendation 2.1, "Seismic." In reaching this determination, staff confirmed the licensee's conclusion that the licensee's ground motion response spectrum (GMRS) exceeds the SSE over the frequency range of 6 to 100 Hz. By letter dated October 27, 2015 [Reference 51], the staff informed the licensee that the expected submittal date for the MNS SPRA is December 31, 2019.

By letter dated December 17, 2014, the licensee submitted its expedited seismic evaluation process (ESEP) report [Reference 44]. In the report, the licensee identified near-term modifications needed to various components in both Units 1 and 2. Further, the licensee is planning more detailed risk evaluations. By letter dated February 4, 2016 [Reference 51], the licensee notified the NRC that the near-term modifications had been completed. By letter dated March 17, 2016 [Reference 53], the NRC staff completed its review of the ESEP report and stated that the licensee's implementation of the interim evaluation met the intent of the guidance.

As the licensee's seismic reevaluation activities are completed, the licensee will enter appropriate issues into the corrective action program. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.2 Flooding

In its FIP, the licensee described that the limiting site flooding event for MNS is the Probable Maximum Precipitation (PMP) event, which is of limited duration and water level. As described in UFSAR Sections 2.4, 2.4.10, and 3.4, the current design-basis is that MNS Seismic Category I structures are not susceptible to external flooding from the PMP or Probable Maximum Flood (PMF) Events.

The licensee has submitted its FHRR [Reference 21]. The flood reevaluation considered the appropriate mechanisms and combined effect floods as described in the 50.54(f) letter. In its FHRR, the licensee described interim actions to preclude maximum water surface elevations from resulting in flooding of plant buildings important to nuclear safety. The NRC summarized the results of the FHRR review in the Interim Staff Response issued by letter dated September 3, 2015 [Reference 46], and concluded that the current design-basis flood levels are exceeded for the following potential flood mechanisms: local intense precipitation, flooding in reservoirs, Lake Norman storm surge, and failure of dams and onsite water control structures due to combined effect flooding. In the Interim Staff Response letter the NRC staff also concluded that the licensee can use the FHRR summary results as suitable input to its reevaluated flood hazard mitigating strategies assessment (MSA). The MSA will be performed using the guidance described in NEI 12-06, Revision 2, Appendix G [Reference 55]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 56]. The licensee's MSA will evaluate the mitigating strategies described in this safety evaluation using the revised flood hazard information and, if necessary, make changes to the strategies or equipment. The licensee will submit the MSA by December 2016 for NRC staff review.

The licensee will also complete the integrated assessment requested in the NRC's 50.54(f) letter by performing either a focused evaluation or an integrated assessment documenting hazard response at the site for flood causing mechanisms that are not bounded by the current design basis, as described in the interim hazard letter. Guidance describing suitable methods to complete a focused evaluation or integrated assessment was submitted to the NRC by NEI in NEI 16-05, "External Flooding Integrated Assessment Guidelines," [Reference 55]. In a recent Federal Register notice [Reference 56], the NRC proposed endorsing NEI 16-05, with clarifications. The licensee is expected to complete the focused evaluation by June 2017, or an integrated assessment by December 2018 if necessary, consistent with the guidance described in NRC's COMSECY-15-0019 [Reference 52].

As the licensee's flooding reevaluation activities are completed, the licensee will enter appropriate issues into the corrective action program. The licensee has appropriately screened in this external hazard and identified the hazard levels to be evaluated.

3.5.3 High Winds

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

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

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, Rev. 2, February 2007; if the recommended

tornado design wind speed for a 1E-6/year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located at 35° 25' 59" North latitude and 80° 56' 55" West longitude. In NEI 12-06, Figure 7-1, Contours of Peak-Gust Wind Speeds, Annual Exceedance Probability of 1E-6/year, the location of MNS has a hurricane peak-gust wind speed of 150 mph. In NEI 12-06 Figure 7-2, Recommended Tornado Design Wind Speeds for the 1E-6/year Probability Level indicates the site is in Region 1, which corresponds to a recommended tornado design wind speed of 172 mph. Therefore, the plant screens in for an assessment for high winds from hurricanes and tornados, including missiles produced by these events.

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

3.5.4 Snow, Ice, and Extreme Cold

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

From UFSAR Table 2.9, McGuire Nuclear Station – Vicinity Climatology, the lowest, average daily minimum temperature on a monthly basis is 32.3 °F and occurs in January with the record lowest temperature of -5 °F occurring in both December and February. The maximum 24-hour snowfall was 14.0 inches, occurring in February. UFSAR Section 2.3.1.3, Severe Weather, describes that winter conditions as a rule are not conducive to the development of major snow storms. Long-term records for the area show that the highest 24-hour snowfall was about 18 inches (Winston-Salem, N.C., December, 1930). Ice storms, a much more frequent occurrence, do effect 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° 25' 59" North latitude and 80° 56' 55" West longitude. In addition, the site is located within the region characterized by EPRI as ice severity level 4 (NEI 12-06, Figure 8-2, Maximum Ice Storm Severity Maps). Consequently, the site is subject to severe icing conditions that could cause severe damage to electrical transmission lines. The licensee concludes that the plant screens in for an assessment for snow, ice, and extreme cold hazard. In its FIP, the licensee stated that FLEX equipment is protected from severe temperatures.

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 2.9, McGuire Nuclear Station – Vicinity Climatology, the highest, average daily maximum temperature on a monthly basis is 88.8 °F and occurs in July with the record highest temperature of 104 °F occurring 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 MNS has implemented the diverse FLEX storage option utilizing buildings designed to or evaluated equivalent to American Society of Civil Engineers (ASCE) standard ASCE 7-10, Minimum Design Loads for Buildings and Other Structures. In its FIP, the licensee stated that MNS has three FLEX storage buildings that are 60 feet by 120 feet with multiple access doors. Design requirements for each building include the following:

- Conforms to ASCE 7-10 for seismic ruggedness (greater than two times the SSE)
- Rated to withstand wind loads to greater than 200 mph, which conforms to ASCE 7-10
- Located to provide protection from the design basis flood hazard
- Separated from the other FLEX Buildings by more than 1,200 feet to minimize the potential for multiple buildings to be damaged by tornado missiles
- Includes power for FLEX equipment block heaters and FLEX equipment battery chargers
- Provides severe temperature protection for FLEX equipment

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

3.6.1.1 Seismic

In its FIP, the licensee stated that the FLEX storage buildings conform to ASCE 7-10 for seismic ruggedness (greater than two times the SSE). During the audit, as documented in Reference 17, the licensee stated that their plan is to strap FLEX components in the FLEX buildings to anchor bolts embedded in the floor of each facility to prevent movement during a seismic event.

3.6.1.2 Flooding

In its FIP, the licensee stated that the three FLEX storage buildings are located to provide protection from the design basis flood hazard. The licensee further stated that flooding does not impact deployment paths from the FLEX buildings to the power block.

3.6.1.3 High Winds

In its FIP, the licensee stated that the three FLEX buildings are separated from each other by more than 1,200 feet to minimize the potential for multiple buildings to be damaged by tornados. During the audit, as documented in Reference 17, the licensee provided a detailed analysis to justify placing all three FLEX buildings at least 1400 feet apart from each other. The NRC staff reviewed the licensee's justification in calculation MCC-1512.00-00-0005 and a diagram of the National Oceanic and Atmospheric Association archive tornado data for McGuire (1950-2010) and concluded that having the three FLEX buildings in diverse locations is adequate to avoid damage to multiple buildings during a tornado.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

In its FIP, the licensee stated that the three FLEX buildings include power for FLEX equipment block heaters and FLEX equipment battery chargers and provide severe temperature protection for FLEX equipment.

During the audit, as documented in Reference 17, the licensee provided further information on the FLEX building's capabilities to withstand high temperatures. The staff reviewed the licensee's information and was able to verify during the on-site audit walkdown that the licensee adequately considered high temperatures as part of the design for the FLEX buildings.

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.

In its FIP, the licensee states that redundant equipment is stored in the three FLEX Buildings such that if any single building were destroyed by the BDBEE (e.g., a tornado), sufficient FLEX equipment would remain intact and available for deployment from the remaining two buildings. Specifically, regarding major FLEX equipment, the licensee states that MNS has three portable low-pressure, medium-pressure, and high-pressure pumps, satisfying the N+1 inventory requirement. The licensee further states that MNS has six submersible FLEX sump pumps to address internal flooding, with each of the three FLEX Buildings containing two pumps. Also described are Boggs Boxes, which are portable units that spray water into the SFP. Boggs Boxes are stored in the FLEX Storage Buildings and MNS has three Boggs Boxes, which satisfies the N+1 requirement for equipment redundancy.

Regarding FLEX power generation, in its FIP the licensee states that each of the three FLEX buildings at MNS contains a 600 Vac FLEX DG, and associated power distribution panels (PDPs) and cabling.

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

3.6.3 Conclusions

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

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee stated that the MNS FLEX response strategies plan for deployment of pumps, DGs, and other equipment from the FLEX buildings to locations at the power block to support the various FLEX capabilities.

3.7.1 Means of Deployment

Regarding FLEX equipment transport and debris removal, in its FIP the licensee states that MNS has a Caterpillar 924K front end loader and a Dodge RAM 5500 diesel truck with stake body and pintle hitch for towing of FLEX equipment. These vehicles will be stored in separate FLEX buildings. In addition to the CAT 924K and the diesel truck, MNS Site Services also has other heavy equipment (e.g., tractors, backhoes, skid steers) in diverse locations that can support debris removal and deployment of FLEX response equipment. These vehicles are capable of clearing storm debris or ice/snow following a severe weather event, or rubble which blocks vehicle access to the needed equipment staging locations following a seismic event.

3.7.2 Deployment Strategies

Regarding FLEX deployment for core cooling strategies, in its FIP the licensee stated that the FLEX medium-pressure pumps may be used to deliver feedwater to the SGs as a contingency to the TDAFW pump, and will eventually use it to allow cooldown in order to transition to use of the residual heat removal system. One FLEX medium-pressure pump is stored in each of the FLEX buildings. One FLEX medium-pressure pump is needed for each unit, which will be staged outside near the stairs to the EDG roof and exterior doghouse. Operators will deploy hoses from that location to tie into an appropriate water supply (e.g., 5-inch hose deployed from the FLEX low-pressure booster pump taking suction from the SNSWP). Operators will also deploy hoses from the FLEX medium-pressure pump discharge to one of the FLEX connections for SG makeup.

The FLEX RCS makeup strategy relies on FLEX high-pressure pumps to deliver water to the RCS. One FLEX high-pressure pump is stored in each of the FLEX equipment storage buildings. One FLEX high-pressure pump will be deployed for each unit. The two pumps may be staged in one of three locations outside the auxiliary building. Operators will deploy hoses from the pump suction to the FLEX piping connection on the FWST supply line to provide a suction source for the pump. Operators will deploy hoses from the pump discharge to one of the FLEX connections for RCS makeup.

Regarding portable power supplies, in its FIP the licensee stated that each of the three FLEX buildings at MNS contains a 600 Vac FLEX DG, and associated power distribution panels (PDPs) and cabling. Two of the three FLEX DGs are needed for a dual-unit ELAP event. The licensee has identified six candidate locations around the power block for potentially staging the FLEX DGs. After the DGs are positioned in the selected locations, operators will set up PDPs and deploy cabling to align the DGs to in-plant motor control centers (MCCs). Four PDPs will be set up for each FLEX DG, with two of the PDPs connected directly to the DG and two other PDPs jumpered to the directly-connected PDPs. Each unit has several permanent modified FLEX electrical connections and several other locations that can use portable FLEX MCC buckets deployed from the FLEX buildings. During Phase 3, two NSRC-delivered 4160 Vac 1 MW DGs per unit will connect to an NSRC 4160 Vac distribution center to re-energize one 4KV essential bus. The NSRC distribution center will be connected to the MNS 4160 Vac switchgear. For Unit 1, the preferred deployment location for the two NSRC DGs is the northwest corner of the Auxiliary Building, just north of the Unit 1 SFP roll-up door. For Unit 2, the preferred location is the northeast corner of the Auxiliary Building near the entrance to the hot tool room.

In its FIP, the licensee stated that the MNS FLEX response strategies rely on distribution of raw water from the SNSWP (the UHS) when other sources of water are no longer available. MNS has one FLEX low-pressure pump with 3,000 gpm capacity and two other FLEX low-pressure pumps with 1,500 gpm capacity each. If the 3,000 gpm pump is available following the BDBEE, operators will preferentially deploy that pump rather than the two 1,500 gpm pumps. MNS will connect the discharge of the FLEX low-pressure pump(s) to the fire protection system if that system is available following the BDBEE. If this option is not available, operators can establish raw water distribution from the SNSWP to the station yard using hoses only. Hoses would be routed through the north vehicle access portal (VAP) or the south VAP. If both VAPs are available, Unit 1 may use the south VAP and Unit 2 may use the north VAP to minimize the

length of hose runs. Operators will install Y-connectors on the deployed hose at pre-determined locations to provide access to the various water demands of the FLEX response strategies.

The licensee concluded that ice in the SNSWP will not affect the ability to provide water for FLEX response strategies. The normal intake from the SNSWP is approximately 40 feet under water and there are two independent, safety-related water sources. Implementation of the Phase 2 strategy requires placing suction hoses into the SNSWP. The debris removal equipment will be able to remove any ice at the SNSWP to enable deployment.

In the FIP, the licensee stated that flooding does not impact deployment paths from the FLEX buildings to the power block. As discussed in UFSAR Section 2.5.4.8, soils beneath MNS are not considered susceptible to seismic liquefaction. Therefore, deployment routes will not be affected by seismic liquefaction.

3.7.3 <u>Connection Points</u>

3.7.3.1 <u>Mechanical Connection Points</u>

Core Cooling (SG) Primary and Alternate Connections

The licensee described in its FIP the primary connection for SG makeup as being the feedwater tempering lines. The licensee indicated that these lines were utilized previously for SG makeup for 10 CFR 50.54(hh) requirements and are located on top of the diesel buildings. These lines are protected from high wind and tornado missiles by surrounding robust structures and they connect to all four SGs to allow for symmetric cooldown. The licensee described in the FIP the alternate SG connection as AFW connections that are located in the Interior and Exterior SG doghouse structures. These locations are inside Seismic Category I structures which protects them from all applicable hazards. The licensee described that one connection is provided in each doghouse, and each of the two connections is capable of feeding two SGs. The set of connections provides simultaneous, parallel makeup for all four SGs, thereby supporting symmetric cooldown of the RCS.

RCS Inventory Control Primary and Alternate Connections

The licensee described in its FIP the primary connection for RCS makeup as the addition of borated water to the RCS using the safety injection pump discharge piping on the 750 ft. elevation of the auxiliary building. The licensee also indicated that the suction takes place from a connection on the FWST recirculation header, which is also on the 750 ft. elevation in the auxiliary building. The discharge and suction connections are located on safety-related piping and are within a Seismic Category I structure, which protects them from the applicable external hazards. The licensee described the alternate RCS makeup connection as the addition of borated water to the RCS through the safety injection pump discharge piping located on the 733 ft. elevation of the auxiliary building, which is safety-related and located within a Seismic Category I structure which protects it from the applicable external hazards.

Instrument Air Primary and Alternate Connections

The licensee described in its FIP the primary connection for providing compressed air to the SG PORVs and the TDAFW pump FCVs for Phase 2 by using the instrument air system blackout

header located in the Exterior Doghouses. The licensee described these connections as being located on seismically rugged piping and within a Seismic Category I structure, which protects them from the applicable external hazards. The licensee also described similar connections for the alternate connection except the location would be for the Interior Doghouses with similar protection from the applicable external hazards. The licensee also indicated that the FLEX Phase 1 instrument air system supply uses the FLEX air tanks.

SFP Makeup Primary and Alternate Connections

The licensee described in its FIP the primary strategy SFP makeup, which is deployment of a hose to the SFP deck for delivering water directly to the SFP. The hose is connected to a Boggs Box, which is a portable unit that sprays water into the SFP. The licensee indicated that three Boggs Boxes are stored in the FLEX storage buildings and would be deployed early for an ELAP event. The licensee described the alternate SFP makeup strategy as using the SFP cooling pump suction piping in each Unit. This piping connection is located in the auxiliary building, which is a Seismic Category I structure, and leads through the SFP area. The licensee indicated that all portions of the piping are protected from the applicable external hazards.

3.7.3.2 Electrical Connection Points

Electrical connection points are only applicable for Phases 2 and 3 of the licensee's mitigation strategies for a BDBEE. During Phase 2, the licensee has developed a primary and alternate strategy for supplying power to equipment required to maintain or restore core cooling, containment, and SFP cooling using a combination of permanently installed and portable components. There are three portable FLEX 600 Vac DGs provided for the strategy. One is stored in each of the three FLEX buildings along with its associated PDPs and cabling. The licensee identified six locations to deploy and stage the 600 Vac FLEX DGs. The primary locations are near the auxiliary building and the alternate locations are near the turbine building. The primary locations are the northeast side of auxiliary building at the entrance to the hot tool room, northwest side of auxiliary building near the entrance to the fire brigade equipment storage room, and the north center side of the auxiliary building near the entrance to the RP laundry area, respectively. The alternate locations are near the Unit 1 turbine building northwest rollup door, the Unit 2 turbine building northeast rollup door, and south center of the SB truck door, respectively.

Once the FLEX DGs are positioned in the selected locations, the licensee will set up PDPs and deploy cabling to connect the FLEX DGs to in-plant MCCs. Four PDPs will be set up per FLEX DG, with two of the PDPs connected directly to the FLEX DG and two other PDPs connected to the directly-connected PDPs. The primary connection strategy for the FLEX DGs uses permanently-installed 600 Vac modified MCC buckets with external power connectors to provide power to various electrical loads used in the FLEX strategy. The alternate connection strategy will require the use of portable MCC buckets with external power connectors deployed from the FLEX buildings and installed in dedicated spare breaker locations. The primary connection strategy for Unit 1 utilizes MCCs 1EMXH1, 1EMXA4, 1EMXA2, and 1EMXB4. The alternate connection strategy for Unit 1 utilizes MCCs 1EMXC, 1EMXA3, 1EMXD, and 1EMXB5. The primary connection strategy for Unit 2 utilizes MCCs 2EMXA2, 2EMXA4, and 2EMXB4. The alternate connection strategy for Unit 2 utilizes MCCs 2EMXC, 2EMXA3, 2EMXD, and 2EMXB5. Guideline FG/0/A/FLEX/FSG-20, "FLEX Electrical Distribution," Rev. 1, provides direction for

repowering MCCs and ensuring proper phase rotation before attempting to power equipment from the 600 Vac FLEX DG.

For Phase 3, two NSRC 4160 Vac 1 MW turbine generators per unit will be connected to an NSRC 4160 Vac distribution center to re-energize one 4160 Vac essential bus. The NSRC distribution center will be connected to the MNS 4160 Vac switchgear. For Unit 1, the preferred deployment location for the two NSRC 4160 Vac turbine generators is the northwest corner of the auxiliary building, just north of the Unit 1 SFP roll-up door. For Unit 2, the preferred location is the northeast corner of the auxiliary building near the entrance to the hot tool room. The two NSRC 480 Vac 1 MW turbine generators (1 per unit) will be deployed in the vicinity of the Phase 2 portable FLEX DGs along with a 480 Vac to 600 Vac step-up transformer. The 480 Vac NSRC generators come with the same style and size connectors as the on-site Phase 2 FLEX DGs. Therefore, the licensee's Phase 3 strategy could utilize the Phase 2 electrical connections if needed. Guideline FSG-23, "Long Term FLEX Strategies," provides direction for ensuring proper phase rotation before attempting to power equipment from the 4160 Vac turbine generators.

3.7.4 Accessibility and Lighting

In its FIP, the licensee stated that battery-powered plant emergency lighting and portable lighting (e.g., headlamps, flashlights) for personnel are available for limited durations. FLEX equipment controls are illuminated by light systems incorporated into the equipment skids. Lighting strings powered from the FLEX electrical distribution system are also included in FLEX response equipment for general area lighting in the control room, AFW pump rooms, and auxiliary building at elevations 760 ft., 750 ft., 733 ft., 716 ft., and 695 ft. Portable lighting in other plant areas that lose power can be supplied by FLEX utility power, which is provided by 120 Vac transformers and spider boxes powered from the FLEX DGs. During response to an ELAP, operators will evaluate establishing temporary lighting in the control room, the motor generator set rooms, the battery room, the interior and exterior doghouses, the technical support center (TSC), and the electrical penetration rooms.

3.7.5 Access to Protected and Vital Areas

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

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee stated that MNS has two sources of diesel fuel oil (DFO) for the FLEX response strategies: (1) the MNS garage underground diesel fuel tank, and (2) the four safety-related 50,000 gallon diesel fuel oil storage tanks (DFOSTs) for the plant emergency diesel generators (EDGs). McGuire also has a portable diesel-powered fuel oil transfer skid, which is attached to the DFOST recirculation pump suction line to pump oil from the DFOSTs. In addition, at least one station fuel oil truck with underground tank draft capability is staged for emergency response in the event of severe weather. The MNS analysis shows that the total estimated DFO consumption is 3,600 gallons per day, so the DFOST inventory would be sufficient for several weeks. Operators would obtain additional fuel from off-site sources during Phase 3, if necessary.

During the audit the licensee stated that refueling requirements for FLEX equipment will be a function of usage and frequency and is therefore not predictable. A refueling timeline for major FLEX components has been developed and placed in the FLEX Program Document.

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 McGuire Nuclear Station 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 the Pooled Equipment Inventory Company (PEICo) and AREVA Inc. to provide FLEX Phase 3 management and deployment plans through contractual agreements with every commercial nuclear operating company in the United States.

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

By letter dated September 26, 2014 [Reference 23], the NRC staff issued its assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the staff concluded that SAFER has procured equipment, implemented appropriate processes to maintain the equipment, and developed plans to deliver the equipment needed to support site responses to BDBEEs, consistent with NEI 12-06 guidance; therefore, the staff concluded in its assessment that licensees can reference the SAFER program and implement their SAFER response plans to meet the Phase 3 requirements of Order EA-12-049.

In its FIP, the licensee stated that the SAFER Response Plan for MNS 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.

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) utilized for receipt of ground transported or airlifted equipment from the NSRCs. From Staging Areas C and/or D, the

SAFER team will transport the Phase 3 equipment to the on-site Staging Area B for interim staging prior to it being transported to the final location in the plant (Staging Area A) for use in Phase 3. In its FIP, the licensee stated that for MNS, Alternate Staging Area D is not used. Staging Area C is the Kings Mountain Training Center. When MNS is ready to receive equipment, SAFER equipment will then be delivered to Staging Area B, which is an overflow parking lot at the MNS site near FLEX Building 2. The licensee has identified primary and alternate driving routes from Staging Area C to Staging Area B. The licensee will coordinate with the state of North Carolina to determine the condition of bridges along the travel path. If road travel from Staging Area C to Staging Area B cannot be accomplished, then Staging Area B will receive SAFER equipment directly via helicopter airlift. The licensee identified two access routes from Staging Area B into the protected area with the primary access being through the normal Vehicle Access Portal (VAP) on the eastern side of the site, and the secondary access point being on the south west side of the site.

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 MNS, 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 expected following a BDBEE resulting in an ELAP with loss of normal access to the ultimate heat sink.

The primary concern with regard to ventilation is the heat buildup which occurs with the loss of forced ventilation in areas that continue to have heat loads. The licensee performed several loss of ventilation analyses to assess the potential impacts of temperatures expected in specific areas related to FLEX implementation to ensure the environmental conditions remain acceptable for personnel habitability and within equipment qualification limits.

The key areas identified for all phases of execution of the FLEX strategy activities are the Main Control Room (MCR), TDAFW Pump Rooms, Vital Battery Rooms, and Electrical Penetration and Switchgear Rooms. The licensee evaluated these areas to determine the temperature profiles following an ELAP with loss of normal access to the ultimate heat sink event. The NRC staff reviewed calculation MCC-1240.00-00-0010, "FLEX Mitigation Strategy: Room Heatup Evaluation for Auxiliary, Service, and Fuel Buildings during ELAP Event," Rev. 1.

In its FIP, the licensee states that they plan to open the MCR doors within 3 hours of the start of the event to maintain temperature at an acceptable level during Phase 1. The licensee also plans to deploy at least 4 spot coolers and 8 FLEX HVAC units to the MCR to maintain

temperature below 120 °F. Based on temperatures remaining below 120 °F (the temperature limit, as identified in NUMARC-87-00, for electronic equipment to be able to survive indefinitely), the 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.

Calculation MCC-1240.00-00-0010 evaluated two cases for heat-up of the TDAFW pump room. The first case determined that, with no door opening or mitigating actions taken, the maximum temperature reached was 110 °F at the end of the 120-hr period of analysis. Furthermore, the second case demonstrated that, with doors opened at 6 hours and a fan placed at 10 hours following an ELAP-initiating event, the maximum temperature at 120 hours was reduced to 103.5 °F. The administrative controls to govern these mitigating actions are contained in FSG-5.

The Vital Battery Room analysis contained in calculation MCC-1240.00-00-0010 also 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 5 hours after the ELAP-initiating event, the maximum temperature was shown to be reduced to 117 °F. Guideline FSG-05 provides direction for opening battery room doors and establishing portable ventilation in the vital battery rooms. The calculation also showed a Phase 3 strategy of repowering the annulus ventilation fans at 96 hours to reduce and maintain the battery room temperature to 104 °F. Guidelines FSG-12, "Alternate Containment Cooling" and FSG-23, "Long Term FLEX Strategies" provide direction for opening doors and starting annulus ventilation fans when the essential bus is re-powered by the NSRC FLEX turbine generators.

For the Electrical Penetration Rooms and Switchgear Rooms, calculation MCC-1240.00-00-0010 determined that, even with no mitigating actions taken, the temperature in each of the respective rooms and elevations would not exceed 110 °F.

Guideline FSG-05 provides instructions to open doors and set up fans as needed to enhance cooling. Based on temperatures remaining at or 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," Rev. 1, for electronic equipment to be able to survive indefinitely), the NRC staff finds that the equipment in the MCR, TDAFW Pump Rooms, Electrical Equipment Rooms, Battery Rooms, and Switchgear Rooms will not be adversely impacted and should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP with loss of normal access to the ultimate heat sink event.

3.9.1.2 Loss of Heating

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 to initiate borated makeup to the RCS within 13 hours as indicated in the FLEX timeline would provide the flow 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 doghouses would be subject to freezing. The 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. The NRC staff verified during the audit that the piping and instrumentation as described are located in areas that will not require heat tracing for cold weather conditions.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

The staff reviewed MNS calculation, MCC-1612.00-00-0006, "Control of Hydrogen Generated by Station Batteries," Rev. 1, 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. The licensee stated in the FIP that hydrogen generation and build-up in an individual battery room enclosure will remain below 2 percent for at least 13 hours of battery charging. The individual battery room enclosures are part of an overall battery compartment, which will remain below 2 percent for at least 15 days. The licensee plans to open doors to the battery rooms after battery charging is initiated, and deploy small portable fans within 12 hours of commencing battery charging to circulate air in the battery rooms and prevent excessive hydrogen gas accumulation. The licensee also stated in its FIP that the ventilation capacity required to maintain acceptable hydrogen concentration is 1.1 ft^3/min and the portable fans used in the FLEX strategy will provide ventilation flow far in excess of this minimum requirement.

Based on the evaluation above, the NRC staff concludes that hydrogen accumulation in the MNS safety-related battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP, because the licensee plans to open battery room doors and provide portable ventilation during Phase 2.

3.9.2 Personnel Habitability

To address room heat-up concerns during an ELAP, the licensee performed calculation MCC-1240.00-00-0010, "FLEX Mitigation Strategy: Room Heat-up Evaluation for Auxiliary, Service, and Fuel Buildings during ELAP Event."

3.9.2.1 Main Control Room

As stated in the FIP and demonstrated in calculation MCC-1240.00-00-0010, the doors will be opened on the MCR and at least 4 spot coolers deployed to reduce and maintain temperatures at about 80 °F. Administrative controls for opening of the MCR doors is contained in ECA-0.0, "Loss of All AC Power", Rev. 37. Additionally, FSG-5, "Initial Assessment and FLEX Equipment Staging", Rev. 1, discusses deployment of the spot coolers and HVAC units. 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 MCR temperature below the conservative limit for control room 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 the exterior roll-up door to allow steam to escape from the SFP area approximately 18 hours following an ELAP-initiating event. The conditions for different locations on the refueling floor are calculated in MCC-1240.00-00-0011, "FLEX Mitigation Strategy: Room Heat-up Evaluation for Auxiliary, Service, and Fuel Buildings during ELAP Event, Volume 2", Rev. 0. The FIP states that the nominal minimum time to boil for a non-outage situation is approximately 35 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 calculation MCC-1240.00-00-0010, "FLEX Mitigation Strategy: Room Heat-up Evaluation for Auxiliary, Service, and Fuel Buildings during ELAP Event," 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

Identification of credited water sources for the FLEX response strategies is included in the previous sections for each individual strategy.

In its FIP, the licensee stated that as part of the initial assessment of plant systems following a BDBEE, operators will determine the condition of the following water sources:

- Refueling Water Storage Tank (FWST) (one per unit)
- Auxiliary Feedwater Storage Tank (CAST) (one per unit)
- Auxiliary Feedwater Condensate Storage Tank (CACST) (one per unit)
- Standby Nuclear Service Water Pond (SNSWP) (UHS) (serves both units)

- Lake Norman (serves both units)

The licensee also stated that other water sources (e.g., reactor makeup water storage tanks (RMWSTs), recycle holdup tanks (RHTs)) will be assessed as necessary during event response.

The FWST and SNSWP are robust sources.

3.10.1 Steam Generator Make-Up

In its FIP, the licensee stated that for SG makeup, operators will provide water from any of the following sources:

- Auxiliary Feedwater Storage Tank (default)
- Auxiliary Feedwater Condensate Storage Tank
- Condenser Cooling Water system piping embedded volume
- Lake Norman (via Nuclear Service Water and Condenser Cooling Water system piping)
- SNSWP (via Nuclear Service Water and Condenser Cooling Water system piping)

The licensee stated the following in its FIP:

> The embedded RC [condenser cooling water] system captured volume and the SNSWP are the credited sources of water because of their robustness to the applicable hazards. Lake Norman may not be available as a water source (e.g., if there is a dam failure). The CAST and CACST are not protected from external hazards. These tanks are normally aligned as a TDCAP [TDAFW pump] suction source, but automatic realignment of TDCAP suction to embedded RC system captured volume is provided if the CAST and CACST are lost. The CAST and CACST have condensate grade water that will not foul the SGs. If MNS switches to Lake Norman or the SNSWP, raw water is acceptable for use for a limited duration. In this case, water purification equipment from the NSRC will be deployed to establish a clean water source.

The licensee has further calculated the acceptable time duration for use of raw water in the SGs to be at least 12 days. The licensee's FLEX strategy is to be on RHR cooling by 6 days after event initiation, which provides ample margin. If necessary, water purification equipment from the NSRC could be placed into service when it becomes available in Phase 3.

3.10.2 Reactor Coolant System Make-Up

In its FIP, the licensee stated that for RCS inventory control during Phase 2, the FWST is the source of borated makeup water. The licensee stated that the FWST inventory can be replenished using one or more of the following options:

- Boric Acid Tanks (BATs)
- Blended makeup from the BAT and another source (e.g., NSRC-supplied water purification unit, CAST, or raw water
- NSRC-supplied mobile boration skid
- Opposite unit's FWST
- RHTs
- Portable FLEX drop tanks mixing boron and RMWST inventory or raw water

For RCS boration during Phase 2, operators will provide borated water from both of the following sources:

- RCS Cold Leg Accumulators
- FWSTs

The licensee further stated that the SNSWP provides a robust water source that can be credited for long-term FWST makeup for all applicable hazards. However, a clean water source is the preferred option for mixing borated water and refilling the FWST. Tanks containing clean water or the NSRC-supplied water purification equipment would be used rather than raw water, if available.

3.10.3 Spent Fuel Pool Make-Up

In its FIP, the licensee stated that for inventory control of the SFP, operators use raw water through the fire protection system or from hoses, both of which ultimately are pressurized with raw water. The credited source of this water is the SNSWP. The SNSWP will be available following the applicable external hazards. During Phase 3, operators 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 stated that the ice condenser containment design helps maintain containment conditions in all phases of the ELAP event, until the ice bed inventory is depleted. McGuire 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 MNS FLEX response strategies.

3.10.5 Conclusions

Based on the evaluation above, the NRC staff concludes that the licensee has developed guidance that, if implemented appropriately, should maintain satisfactory water sources

following a BDBEE consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, and should adequately address the requirements of the order.

3.11 Shutdown and Refueling Analyses

Order EA-12-049 requires that licensees must be capable of implementing the mitigation strategies in all modes. In general, the discussion above focuses on an ELAP occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. When the ELAP occurs with the plant at power, the mitigation strategy initially focuses on the use of the steam-driven 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 vessel 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, as described in its FIP, shows that following a full core offload to the SFP, about 63 hours are available to implement makeup before boil-off results in the water level in the SFP dropping far enough to uncover fuel assemblies, and the licensee has stated that they have the ability to implement makeup to the SFP within that time.

When a plant is in a shutdown mode in which steam is not available to operate the TDAFW 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 if there is fuel in the reactor vessel. On September 18, 2013, NEI submitted to the NRC a position paper entitled "Shutdown/Refueling Modes" [Reference 36], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 37], 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 27, 2014 [Reference 13], the licensee informed the NRC staff of its plans to follow the guidance in this position paper.

Based on the licensee's incorporation of the guidance from the NRC-endorsed position paper into its FLEX program, 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 FLEX Support Guidelines (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 Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs). 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 EOP 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 the following in regard to the training program:

> 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.

Based on the description provided above, the NRC staff concludes that the licensee's established procedural guidance meets the provisions of NEI 12-06, Section 11.4 (Procedure Guidance). Similarly, the NRC staff concludes that the training plan, including use of the SAT process for the groups most directly impacted by the FLEX program, meets the provisions of NEI 12-06, Section 11.6 (Training).

3.13 Maintenance and Testing of FLEX Equipment

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 [Reference 38], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 39], 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 OIP [Reference 10], the licensee stated that equipment associated with FLEX mitigation strategies will be procured as commercial equipment with design, storage, maintenance, testing, and configuration control in accordance with NEI 12-06 Section 11.1. By letter dated August 27, 2014 [Reference 13], the licensee stated that MNS will follow EPRI Report 30002000623 in the development of maintenance and testing programs for equipment acquired in response to Mitigation Strategies Order EA-12-049, or take justified vendor exception, as appropriate.

Based on the above, 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

The licensee and the NRC staff did not identify any alternatives to NEI 12-06, Revision 0.

3.15 Conclusions for Order EA-12-049

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

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 24], the licensee submitted its OIP for MNS in response to Order EA-12-051. By letter dated June 13, 2013 [Reference 25], the NRC staff sent a Request for Additional Information (RAI) to the licensee. The licensee provided a response by letter dated July 11, 2013 [Reference 26]. By letter dated October 28, 2013 [Reference 27], and November 7, 2013 [Reference 50], the NRC staff issued an Interim Staff Evaluation and a supplement, including RAIs, to the licensee. The licensee provided a response by letter dated February 27, 2014 [Reference 29]. By letter dated October 9, 2014 [Reference 17] the NRC issued an audit report on the licensee's progress.

By letters dated August 26, 2013 [Reference 28], February 27, 2014 [Reference 29], August 27, 2014 [Reference 30], February 26, 2015 [Reference 31], and August 26, 2015 [Reference 32], the licensee submitted status reports for the Integrated Plan. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable SFP level instrumentation which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letters dated November 18, 2014 [Reference 49], and December 7, 2015 [Reference 18], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed a SFP level instrumentation system designed by AREVA Americas, Inc. The NRC staff audited AREVA's SFPI design and test results in support of the review of MNS's OIP in response to Order EA-12-051. During the audit, the NRC staff reviewed the vendor's SFPI system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report on September 15, 2014 [Reference 33].

The staff performed an onsite audit to review the implementation of SFP level instrumentation related to Order EA-12-051. The scope of the audit included verification of (a) site's seismic and environmental conditions enveloped by the equipment qualifications, (b) equipment installation met the requirements and vendor's recommendations, and (c) program features met the requirements. By letter dated October 9, 2014 [Reference 17], the NRC issued an audit report on the licensee's progress. Refer to Section 2.2 above for the regulatory background for this section.

4.1 Levels of Required Monitoring

In its OIP, the licensee stated that indication of SFP level will be provided to monitor from the normal water level, about 771 foot elevation (771'), down to the top of the highest point of any fuel rack seated in the SFP. The level instrumentation will provide the capability to monitor SFP level at the three distinct critical levels identified by NEI 12-02 guidance:

- Level 1 - Level adequate to support operation of the normal fuel pool cooling system. The minimum required level to provide adequate pump suction is 769'.
- Level 2 - Level adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck is 756' (10±1 foot above the top of the highest point of any fuel rack seated in the SFP). This level provides substantial personnel radiation shielding that would allow implementation of local SFP makeup strategies for a beyond design bases event.

- Level 3 - Level where fuel remains covered and actions to implement makeup water addition should no longer be deferred is approximately 746' ±1 foot, the highest point of any fuel rack seated in the SFP.

For Level 1, in its letter dated July 11, 2013 [Reference 26], the licensee stated that the normal SFP water level is 771.4' Elevation. The SFP cooling pump suction piping submergence is lost when water level decreases below 767.8' Elevation. Abnormal procedures secure the SFP cooling pump when water level decreases to 2' below normal. Thus, the Level 1 is considered to be 769.4' Elevation.

In its letter dated July 11, 2013 [Reference 26], the licensee also provided sketches depicting the approximate location of the level sensor and the elevations identified as Levels 1, 2 and 3. These sketches depict Level 1 at 769.4' Elevation, Level 2 at 756' Elevation, and Level 3 at 746' Elevation.

The NRC staff noted that the Level 1 designation is adequate for normal SFP cooling system operation and it is also adequate to ensure the required fuel pool cooling pump 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 ft. above the top of the fuel rack. Level 3 designation is the top of the fuel rack.

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

4.2 <u>Evaluation of Design Features</u>

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

4.2.1 <u>Design Features: Instruments</u>

In its OIP, the licensee stated that the SFP level instrumentation will consist of two permanently installed instruments for each SFP, which will provide continuous SFP level indication. The remote level indication range will be specified to support monitoring SFP levels above the minimum allowed Technical Specification 3.7.13, "Spent Fuel Pool Water Level," (e.g. ~769' Elevation or ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks), and the top of the fuel storage racks at 746' Elev. At least one of the instrument channels will provide remote control room indication.

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

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

4.2.2 Design Features: Arrangement

In its OIP, the licensee stated that the level instruments/channels will be installed in diverse locations and physically arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the SFP. The associated cabling, power supplies and indication for each level instrument/channel will be routed separately from each other. Cable routings will be specified to provide reasonable protection from missiles that may result from damage to the structure over the SFP and refuel floor, as applicable. The conduit and cable routing will be determined by the detailed design.

In its letter dated July 11, 2013 [Reference 26], the licensee provided two figures depicting the location of the primary SFPI channel in the SFP area and one figure depicting the location of the back-up SFPI channel in the SFP area. The letter also stated that the wave guided radar pipe is routed slightly above the SFP operating deck, and through a floor core bore penetration down to the sensor electronics located on the 767' Elevation of the Auxiliary Building, and indication/display is provided in the main control room. A local control panel is located adjacent to the sensor electronics located on the 767' Elevation of the Auxiliary Building. The signal cable is routed from the sensor electronics to the cable spread room and to the indication on the main control board 1/2MC9. All associated channel electronics, and power/signal cabling are located in the Seismic Category I Auxiliary Building, and/or Control Room. The back-up SFP level channel is a mechanical pressure gauge and requires no power supply, nor signal cable. The back-up SFP level channel display/read-out is located on the 733' Elevation of the Auxiliary Building in the electrical penetration room.

In its letter dated February 27, 2014 [Reference 29], the licensee further provided several figures depicting the location of primary and back-up SFP level instrument sensors in the SFP area, the primary SFP level instrument wave guided radar piping, the cable routings, and the back-up SFP level instrument impulse tubing route within the reactor building annulus and the local field display location. The letter also stated that the primary SFP level instrument channel and associated cable route layout is completely separated from all components of the back-up level instrument channel by walls and/or floors of the Seismic Category I Auxiliary Building. The primary channel wave guided pipe and horn are mounted to the SFP operating deck floor (~779' Elevation) at the south end of the SFP Building. This location affords reasonable protection from a postulated missile emanating from the north wall of the SFP Building, which was not designed for missile protection. The remaining SFP building walls and ceiling are designed Seismic Category I. The wave guided pipe passes through a penetration in the SFP operating deck floor, and is connected to the electronic sensor/transmitter which is located on the lower elevation (767' Elevation) of the Seismic Category I Auxiliary Building wall. The back-up channel consists of instrument sense-line tubing which is connected to the SFP transfer tube inside the Reactor Building annulus and is routed to a mechanical pressure gauge located on the 733' elevation of the Auxiliary Building. The back-up channel is entirely contained within Seismic Category I structures, and protected from wind driven missiles.

During the onsite audit walkdown, the NRC staff noted that the primary level instrument was located with reasonable protection from potential missile hazards. Although the north wall of the SFP building is not Seismic Category I qualified, the instrument is positioned farthest from that wall and has a large crane structure at the far end of the pool that provides missile protection from the north wall. The back-up level instrument does not have physical exposure on the refueling floor.

While reviewing the figure depicting the layout of the back-up level instrument channel, the NRC staff noted that the back-up channel's centerline of the SFP transfer tube, where the process connection is located, is at elevation of 737' while the instrument level display is at elevation of 733' 4". In ISE RAI-2, the NRC staff requested a description of the back-up level instrument's required slope for the impulse tubing between the process zero reference point and the read-out/display, and how the required slope is maintained. In its letter dated February 27, 2014 [Reference 29], the licensee stated that the Instrumentation and Controls Field Installation Specification outlines instrumentation impulse tubing slope requirements. The specification generally requires a minimum slope of 1/4" per linear foot. Typically if exceptions are taken to the minimum slope requirements, then the design would include provisions for installation of vent/drain valves as applicable. The design of the back-up SFP level channel instrument impulse tubing included an accessible high point vent valve above the display/gauge to facilitate fill and vent. Thus, the impulse tubing slope is judged not to be a concern.

During the onsite audit, the staff reviewed the following:

- Drawing No. MCID-1499-NV.77, "Instrument Details Spent Fuel Pool (KF Sys) Wide Range Level Instrumentation Backup"
- Drawing No. MC-1414-22.20-00, "Piping Layout NV Plan EL. 725'-0" Thru 738'-3" Reactor Building"
- Drawing No. MC-1414-22.20-01, "Piping Layout NV System Sections Reactor Building"

The NRC staff noted that the Drawing MCID-1499-NV.77 specifies that the slope of the impulse line downward from the vent valve to the instrument should be a minimum of ¼" per foot and the vent valve should be located at the highest point. The licensee, during the onsite audit interview, stated that MNS cannot maintain minimum slope of ¼" per foot requirement for the entire impulse line and takes exception to this requirement by installing the vent valve at the highest point. The staff found this acceptable and has verified the installed vent valve in Unit 1 during the walkdown.

The NRC staff had concerns with the effects that outside temperature and borated water within the impulse line and at the process tap on the SFP transfer tube may have on the back-up instrument. In response to the staff's concern, in its letter dated February 27, 2014 [Reference 29], the licensee stated that the back-up SFP level instrument impulse tubing is confined to the lower elevations (725' to 735' Elevation) of the Reactor Building Annulus and Auxiliary Building areas which are below site grade elevation, which is 760' Elevation. The surrounding subterranean ground temperature and Reactor Building concrete thermal inertia would serve to moderate the Reactor Building annulus and Auxiliary Building temperatures due to outdoor temperature extremes. During a postulated BDB event additional annulus heat input would be expected to occur through the containment steel vessel wall from lower containment. Based on

engineering judgment, the relative temperate climate associated with the McGuire site location could not support sustained cold outdoor temperature conditions which could result in freezing of the back-up SFP level instrument sensing line.

The NRC staff found the response adequately addressed the staff's concerns with respect to the instrument sensing line and verified the response by reviewing the following drawings during the onsite audit:

- MCID-1499-NV.77, "Instrument Details Spent Fuel Pool (KF Sys) Wide Range Level Instrumentation Backup" (Unit 1)

- MCID-2499-NV.77, "Instrument Details Spent Fuel Pool (KF Sys) Wide Range Level Instrumentation Backup Channel Flex Strategy" (Unit 2)

- MC-1414-22.20-00, "Piping Layout NV Plan EL. 725'-0" Thru 738'-3" Reactor Building"

- MC-1414-22.20-01, "Piping Layout NV System Sections Reactor Building"

- MC-1091-1, "Reactor Building 1 Concrete Shell Developed Elevation-Conc. EL.722'+6" Thru EL. 875'+4 ½""

- MC-1093-01.00, "Reactor Building 2 Concrete Shell Developed Elevation-Conc. EL.722'+6" Thru EL. 875'+4 ½" Sheet 1"

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

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

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

In its OIP, the licensee stated that permanently installed instruments will be mounted to retain the component design configuration during and following the maximum seismic ground motion considered in the design SFP structure or applicable structure in which the component is located.

In its letter dated February 27, 2014 [Reference 29], the licensee 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 wave guide piping and its components and the seismic accelerations for the building structure. To meet the design criteria for a BDB Event, the loading for the mounting supports were generated using a

minimum of 2xSSE seismic accelerations. The mounting designs for these supports are qualified by calculations using the Manual For Steel Construction AISC 9th Edition, Hilti Product Technical Guides, and site specific specifications. 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. The base plate is anchored to the wall with four (4) concrete anchor bolts. The cantilevered length from the pipe to the wall is 10". The generic sensor mounting support was designed for generic enveloping seismic accelerations of 10g (horizontal) and 6.67g (vertical), which readily envelopes the site seismic response spectra. The calculation further assumed an enveloping sensor cantilevered length.

The horn mounting support is qualified by a site specific calculation using a simple C-channel steel section welded on a ½" thick base plate on the concrete operating deck floor in the SFP area. The base plate is anchored to the floor with four (4) concrete anchor bolts. The cantilevered length from the pipe to the floor is 3". The horn mounting support conservatively uses generic seismic accelerations of 10g (horizontal) and 6.67g (vertical). The actual cantilevered length used in this qualification is 3" from the floor. The intermediate mounting supports are qualified by a site specific calculation. These mounting supports consist of U-bolting the pipe to a steel angle and welding the steel angle to a ¾" thick steel base plate. The base plate is anchored either to the Auxiliary Building concrete wall or to the concrete operating deck floor in the SFP area. The intermediate mounting supports uses the 2xSSE design seismic accelerations of 3.45g (horizontal) and 0.50g (vertical). The mounting of the power control panel consists of bolting the power control panel to two sections of unistrut using 3/8" A307 bolts. Each section of unistrut is anchored to the wall with two (2) concrete anchor bolts.

During the onsite audit, the NRC staff reviewed Calculation MCC-1223.20-00-0022, "Seismic Induced Hydrodynamic Response in the Catawba and McGuire Spent Fuel Pools (Sloshing Analyses)," Revision 1, which evaluated the hydrodynamic response of the SFP to an SSE (design-basis earthquake) response spectrum. The analysis uses 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 the analysis, the water 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 analysis evaluates the flow conditions local to the guided wave instrument to determine whether the SFP water interacts with the instrument resulting in hydrodynamic loads. In its conclusion, the analysis 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, which are well below the 6 foot distance between the SFP liquid surface and the horn. The NRC staff noted that the sloshing analysis' assumptions, analytical model, and conclusion are adequate. The staff also conducted a walkdown to verify the as-built elevations of the waveguide horn and normal SFP water level.

The NRC staff reviewed the mounting specifications for the SFP level instrumentation, including the methodology and design criteria used to estimate the total loading on the mounting devices.

The staff also reviewed the design inputs and the methodology used to qualify the structural integrity of the affected structures and equipment for each of the SFP level mounting attachments. Based on that review, the staff found that the criteria established by the licensee adequately accounted for the appropriate structural loading conditions, including seismic and hydrodynamic loads.

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

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

Appendix A-1 of the guidance in NEI 12-02 describes a quality assurance process for non-safety systems and equipment that 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.

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

4.2.4.2 Equipment Reliability

4.2.4.2.1 Temperature, Humidity, and Radiation

In its OIP, the licensee stated that the level instrumentation shall remain functional and maintain required accuracy capability after exposure to any applicable harsh environmental conditions for the equipment location. The level instrumentation and associated cabling will be specified to be reliable at the maximum temperature, humidity, and radiation levels predicted during an ELAP event at their installed locations.

In its letter dated July 11, 2013 [Reference 26], the licensee further stated that the back-up SFP level channel is a mechanical pressure gauge, and is remotely located from the SFP area. It is not exposed to SFP steam or radiation. The gauge design temperature limits will be suitable for the location environment. The postulated temperature in the spent fuel pool area that results from a boiling pool is 100 °C (212 °F). For the primary channel, the electronics will be located outside of the spent fuel pool room in an area where the temperature will not exceed the radar sensor electronics rated design temperature. The maximum humidity postulated for the spent fuel pool area is 100% relative humidity, saturated steam. The electronics will be located outside of the spent fuel pool room in an area away from the steam atmosphere. The area above and around the pool will be subject to large amounts of radiation in the event water level decreases near the top of the fuel racks. The only parts of the measurement channel in the

pool radiation environment are the metallic waveguide and horn, which are not susceptible to the expected levels of radiation. The sensor electronics will be located in an area that does not exceed their $1x10^3$ rad design limit for the required operating time, or the design will provide shielding as required.

In its letter dated February 27, 2014 [Reference 29], the licensee further stated that formal analyses were performed for more limiting areas of the Auxiliary Building during a postulated ELAP event. The analyses concluded ambient temperatures for the more limiting locations would not exceed 120 °F after 7 days without mitigating cooling actions. Based on engineering judgment, the ambient temperature analyses for these areas would readily bound those associated with the location of the primary SFP level instrumentation electronics. The primary channel sensor and power/control panel are located on the 767' elevation of the Auxiliary Building, in the vicinity of HVAC equipment which would be idle during a postulated ELAP event. The limiting design temperature for the primary channel sensor and power/control panel components is 149 °F, and is judged to be acceptable for the proposed location. The primary channel sensor and power/control panel are located on the 767' elevation of the Auxiliary Building. The ambient humidity in this location during a postulated ELAP event would be expected to be below 100% RH. The power/control panel enclosure is rated NEMA 4X and provides protection to the internal components from the effects of high humidity environments.

The location of the electronics is shielded from the direct shine from the fuel, and bounce and scatter effects above the pool. The sensor is shielded from SFP area dose during a postulated event with SFP level at Level 3 by a 4' thick concrete wall between the SFP and the Auxiliary Building, and by the 3' thick concrete floor below. A location specific dose analysis was performed to determine the expected sensor dose for a postulated BDB event with SFP level at Level 3. The analysis determined the total integrated dose (TID) for a postulated BDB event with SFP at Level 3 to be about 119 Rads over the expected maximum 7 day mission time. The associated BDB dose-rate at the electronics location was calculated to be 0.50 Rad/hr. The electronics in the VEGAPULS 62 ER sensor, displays and power control panel are considered to be qualified for $1x10^3$ Rad.

The NRC staff found the licensee adequately addressed the equipment reliability of SFP level instrumentation with respect to temperature, humidity and radiation. The equipment qualification envelops the expected McGuire plant radiation, temperature, and humidity conditions during a BDB event. The equipment environmental testing demonstrated that the SFP instrumentation should remain functional during the expected BDB conditions.

4.2.4.2.2 Shock and Vibration

In its letter dated February 27, 2014 [Reference 29], the licensee stated that the VEGAPULS 66 Through Air Radar sensor and PLICSCOM indicating and adjustment module mounted to the sensor were shock tested in accordance with MIL-STD-901 D and vibration tested in accordance with MIL-STD-167-1. The shock test consisted of a total of nine (9) shock blows, three (3) through each of the three (3) principal axes of the sensor, delivered to the anvil plate of the shock machine. The heights of hammer drop for the shock blows in each axis were one (1) foot, three (3) feet and five (5) feet. The VEGAPULS 62 ER Through Air Radar sensor has also been shock tested in accordance with EN60068-2-27 (100g, 6ms), ten (10) shock blows applied along a radial line through the support flange. The vibration test frequencies ranged from 4 Hz to 50 Hz with amplitudes ranging from 0.048" at the low frequencies to 0.006" at the higher

frequencies. Additional testing of the VEGA PULS 62 ER sensor was performed in accordance with EN 60068-2-6 Method 204 (except 4g, 200 Hz).

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

4.2.4.2.3 Seismic

In its OIP, the licensee stated that permanently installed instruments will be mounted to retain the component design configuration during and following the maximum seismic ground motion considered in the design SFP structure or applicable structure in which the component is located.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that the back-up SFP level channel is a mechanical pressure gauge that is considered to be seismically rugged. The pressure gauge will be seismically mounted and its reliability is based on the successful operating history for similar type devices.

In its letter dated February 27, 2014 [Reference 29], the licensee further stated that the sensor, indicator, power control panel, horn end of the waveguide, standard pool end and sensor end mounting brackets, and waveguide piping were successfully seismically tested in accordance with the requirements of the IEEE Standard 344-2004. The system was monitored for operability before and after the resonance search and seismic tests. The required response spectra used for the five OBEs and one SSE in the test were taken from EPRI TR-107330, Figure 4-5. This test level exceeds the building response spectra where equipment will be located.

The NRC staff found the licensee adequately addressed the McGuire SFPI equipment's seismic qualification. The staff performed a vendor audit and found the AREVA's SFPI design and qualification process acceptable.

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

4.2.5 Design Features: Independence

In its OIP, the licensee stated that the level instruments and any associated cabling (for each Unit SFP) will be physically separated and electrically independent of one another.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that the primary and back-up SFP level channels employ diverse sensing technology. The primary SFP level channel consists of a wave guided radar pipe and horn sensing assembly located on the SFP operating deck. The back-up SFP level channel is a mechanical pressure gauge that is remotely located from the SFP area and any primary level channel components/cabling.

In its letter dated February 27, 2014 [Reference 29], the licensee further stated that the primary level channel is readily afforded power supply and electrical independence from the back-up channel, due to the fact that the back-up channel requires no power or signal cable.

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

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

4.2.6 Design Features: Power Supplies

In its OIP, the licensee stated that power supplies for each SFP instrument/channel shall be electrically separate. If powered, the level instrumentation shall have provisions for emergency back-up power source such as batteries, which are rechargeable or replaceable. The back-up power source(s) must have sufficient capacity to maintain the level indication function until offsite power or other offsite emergency resources provided by FLEX procedures becomes available, consistent with the guidance of NEI 12-02.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that the primary SFP level channel dedicated battery capacity is based on ability of the sensor to supply full load (20 mA) for the duration specified in the plant FLEX mitigation strategy with built-in safety margin. The preliminary estimate of battery capacity is expected to be at approximately 6-7 days. It is estimated that a minimum battery capacity of 72 hours is required to align with the FLEX mitigation plan. If required, battery replacement provisions will be included in the FLEX Phase III strategy to provide continued SFP level monitoring capability. The back-up SFP level channel is a mechanical pressure gauge that does not require power.

In its letter dated February 27, 2014 [Reference 29], the licensee further stated that the primary level channel power control cabinet provides the normal and back-up battery power supply, and is located adjacent to the sensor. The normal power supply is to be provided by non-essential 120 Vac panel boards located on the 750' and 767' Elevations (Unit 2 and 1, respectively) of the Auxiliary Building. Vendor analyses shows the battery capacity (at 20 mA continuous discharge) can support about 130 hours of operation at -22 °F and about 230 hours at 32 °F.

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

4.2.7 Design Features: Accuracy

In its OIP, the licensee stated that the SFP level instrumentation will be designed to maintain their design accuracy without recalibration following a power interruption or change in power source. Accuracy will consider SFP post-event conditions, e.g., saturated water, steam environment, or concentrated borated water, or applicable limiting environmental conditions for the installed location.

In its letter dated July 11, 2013 [Reference 26], the licensee further stated that the manufacturer reference accuracy for the primary SFP level channel is no greater than ±1 inch based on tests performed by AREVA. This is the design accuracy value that will be specified for the primary SFP level instrument channel. The stainless steel horn antenna and waveguide pipe that is exposed to BDB conditions is unaffected by radiation, temperature and humidity other than a minor effect of condensation forming on the waveguide inner walls that will have a slight slowing effect on the radar pulse velocity. Testing performed by AREVA using saturated steam and saturated steam combined with smoke indicate that the overall effect on the instrument accuracy is minimal. In its letter dated February 27, 2014 [Reference 29], the licensee stated that the total loop uncertainty due to BDB conditions described above was calculated to not exceed ± 10 inches. The licensee also stated that the total loop uncertainty for the back-up SFP level channel was calculated to not exceed ± 12 inches for BDB conditions.

During the AREVA SFPI audit [Reference 33], the NRC staff noted that per AREVA Document No. 51-9220845-000, condensation formed in the waveguide tube can cause degradation of the signal. To address the NRC staff's concern, AREVA proposed installing a horn cover to prevent the intrusion of moisture in the waveguide. During the onsite audit at McGuire, the NRC staff reviewed AREVA document 141-9225014-003, "Horn Cover Installation, Steam Test, and Shear Test," dated July 2, 2014, for the horn cover and found it acceptable.

The NRC staff finds that the licensee adequately addressed instrument channel accuracy. The order requires that the instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration. The back-up SFP level channel is a mechanical pressure gauge which does not have power supplied, therefore there is no effect from loss of power. The licensee calculated that the primary SFP level channel has a total loop uncertainty under BDB conditions of ± 10 inches, and the testing performed indicates that if normal power is lost and power is shifted to the battery backup, there is no significant effect on channel accuracy.

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

4.2.8 Design Features: Testing

In its OIP, the licensee stated that the instrument channel design shall provide for routine testing and calibration. Testing will be consistent with the guidelines of NRC JLD-ISG-2012-03 and NEI 12-02.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that the primary SFP level channel has multi-point testing capability, in that the radar horn antenna can be rotated away

from the SFP water surface and aimed at a movable metal target that is positioned at known distances from the horn. This allows checking for correct readings at various points across the instrument measurement range and validates the functionality of the installed system. In its letter dated December 7, 2015 [Reference 18], the licensee further stated that periodic verification of channel functionality can also be achieved by varying SFP water level (a minimum of two levels) and verifying proper level indication. The back-up SFP level channel design readily supports periodic calibration across its monitoring range. The instrument is to be equipped with a calibration test tee and can be isolated from the process for routine calibrations.

Periodic channel checks will be established for the primary and back-up SFP level channels to verify proper instrument operation. The frequency of the channel checks is expected to be at least monthly (± 25% grace period). Instrument channel calibration frequency will be based on the manufacturer's recommended frequency, and/or as established based on operating experience within the preventive maintenance program. As part of the periodic calibration surveillance for the primary SFP level channel, further functional verifications will be performed to verify proper operation of the battery back-up feature on a simulated loss of normal AC power. The channel checks will be performed by Operations surveillance procedures, and the instrument calibrations will be performed by Maintenance instrumentation calibration surveillance procedures. Model work orders will be established within the periodic maintenance program to govern the scheduling and performance of the periodic calibrations. Routine preventive maintenance required during normal operation is limited to periodic channel calibration, and/or battery replacement.

The NRC staff noted that 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 order requires that the instrument channel design shall provide for routine testing and calibration, and the licensee has shown that the design of both the primary and back-up level channels provides for this.

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

4.2.9 Design Features: Display

In its OIP, the licensee stated that the instrument displays for each SFP level instrument will be provided in the main control room or other accessible location and that the displays will be consistent with the guidelines of NRC JLD-ISG-2012-03 and NEI 12-02.

In its letter dated July 11, 2013 [Reference 26], the licensee further stated that the primary SFP level channel has a local display on the Auxiliary Building on 767' Elevation, and in the main control room. The back-up SFP level channel display is in the electrical penetration room, Auxiliary Building on 733' Elevation. The back-up SFP channel display is located outside of the main control room and remote from the SFP area. The display location is located outside of any locked high radiation areas, and is accessible by operations personnel during a postulated BDB event. The back-up level channel read-out displays are located in Seismic Category I structures, which are protected from potential threats posed by the applicable external hazards, such as flooding, seismic and tornado missiles. Personnel access to the display location relies upon the stairwells which provide access to the Auxiliary Feedwater Pump Rooms and Auxiliary

Shutdown Panels. During a postulated ELAP event ambient temperatures at this location would be not be expected to prohibit periodic personnel access to monitor SFP levels. The estimated time for personnel to access the back-up channel display is 10-15 minutes.

The NRC staff found the licensee adequately addressed the display requirements. The licensee's proposed location for the primary and backup SFP instrumentation displays appears to be consistent with NEI 12-02. The displays are located in seismically qualified buildings and the accessibility of the Main Control Room and Auxiliary Building following an ELAP is considered acceptable.

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

4.3 Evaluation of Programmatic Controls

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

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that the Systematic Approach to Training (SAT) process will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training. Training will be completed prior to placing the instrumentation in service. Station personnel performing functions associated with the SFP level instrumentation will be trained to perform the job specific functions necessary for their assigned tasks.

NEI 12-02 specifies that the SAT process be used to identify the population to be trained, and also to determine both the initial and continuing elements of the required training. Based on the licensee's statement above, the NRC staff finds that the licensee's plan to train maintenance and operations personnel appears to be consistent with NEI 12-02, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.2 Programmatic Controls: Procedures

In its OIP, the licensee stated that station 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 be developed to address strategy to ensure SFP water addition is initiated at an appropriate time consistent with implementation of NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that a new Selected Licensee Commitment (SLC) will be established for the primary and back-up SFP level channels. The SLC will specify the required frequency of performance for periodic channel checks, functional checks, and calibrations, as appropriate. The SLC will outline allowed out of

service time-frames consistent with NEI 12-02 requirements. The SLC will specify required remedial actions, in the event one or more channels cannot be restored operable within the allowed out of service time-frame. The remedial actions will be consistent with NEI 12-02 requirements. Allowed channel out of service time-frames will be tracked by the Technical Specification Action Item Log (TSAIL) program in accordance with Operations Management Procedure 5-3. Operational surveillances will be established to periodically verify proper level channel operation, which will consist of periodic primary and back-up SFP level channel checks. Operations Management Procedure (OMP) 5-3 (Operations Periodic Test Program) governs the requirements for scheduling, reviewing and evaluation of periodic operational tests. The OMP requires unacceptable test results to be documented within the Corrective Action Program.

Preventive maintenance tasks will be established in accordance with Nuclear Operating Fleet Administrative Procedure AD-EG-ALL-1202, which governs the Preventive Maintenance program bases, task planning and scheduling, execution, feedback, and change process. The preventive maintenance tasks will entail periodic level channel calibration, and functional checks. Subsequent to the performance of maintenance activities, post maintenance testing will be performed to ensure the SFP level instrumentation is properly functioning prior to return to service. Work Process Manual 501 and Nuclear Station Directive 408 govern the station requirements for testing. FLEX Support Guides (FSGs), Emergency and/or Abnormal operating procedures will incorporate use of the primary and back-up SFP level instrumentation for monitoring/maintaining SFP inventory for BDB events, as appropriate.

In its letter dated February 27, 2014 [Reference 29], the licensee stated that in concert with the engineering change process, the planned station procedures include the following:

- Existing steps within Emergency and Abnormal Procedures will be modified to allow monitoring of SFP level via the primary and/or back-up SFP level channels. Procedures include the following:

  - ECA 0.0 (Loss of All AC Power)

  - G-1 (Generic Enclosures) (This EP is referenced from other Emergency and Abnormal procedures.)

  - AP/24 (Loss of Plant Control Due to Fire or Sabotage).

  - Flex Support Guide (FSG) for Alternate SFP Make-up and Cooling

- Operations surveillance procedure to periodically verify proper operation of the primary and back-up SFP level instrumentation. The procedure will perform periodic channel checks or comparisons between available SFP level instrumentation to verify proper operation of the primary and back-up SFP level instrumentation. This procedure will also serve to verify proper channel functionality within 60 days of a planned refueling outage, as required by NEI 12-02. The procedure is intended to provide a means of detection of channel drift and/or malfunction.

- Maintenance procedures for periodic calibration of the primary and back-up SFP level instrumentation and functional check of primary channel

battery back-up capability. The procedure(s) will verify proper operation of the level instrumentation, and provide instruction for equipment calibration adjustment within design accuracy requirements.

The NRC staff noted that the licensee adequately addressed the procedure requirements for the testing, surveillance, calibration, and operation of the primary and backup SFP level instrument channels.

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

4.3.3 Programmatic Controls: Testing and Calibration

In its OIP, the licensee stated that testing and calibration of the instrumentation will be consistent with vendor recommendation or other documented basis. Calibration will be specific to the mounted instrument(s) and the display(s). Station procedures and preventive maintenance will be developed to perform required instrumentation maintenance, testing, periodic calibrations, and/or functional checks.

In its letter dated July 11, 2013 [Reference 26], the licensee stated that a new SLC will be established to address allowed out service time-frames and required compensatory actions as follows:

1. For a single primary or back-up SFP level channel out of service beyond 90 days, the compensatory actions could include one or more of the following:

    - Increased surveillance (channel check) to verify functionality of the remaining operable level channel
    - Implement equipment protective measures
    - Increased operator visual surveillance of the SFP level and area
    - Maintain elevated SFP level
    - Reduce SFP temperatures
    - Supplemental operations staffing

2. For both the primary and back-up SFP level channels out of service, the compensatory actions could include one or more of the following:

    - Increased operator visual surveillance of the SFP level and area
    - Maintain elevated SFP level
    - Reduce SFP temperatures
    - Supplemental operations staffing
    - Pre-stage FLEX support equipment (nozzles, hoses, etc) which are relied upon for SFP makeup. Pre-staged equipment would be located within Seismic Category I structures.

In its letter dated February 27, 2014 [Reference 29], the licensee further stated that the SLC will further require the functional testing be performed to verify proper channel operation within 60

days of a planned refueling outage, as required by NEI 12-02. This testing is expected to entail the performance of a channel check, and functional verification of the primary channel battery back-up capability. The channel out of service durations, required remedial actions and required action timeframes will be formally controlled similar to that for Technical Specifications. The corrective action program (CAP) would formally evaluate "functionality" for the SFP level channels and establish appropriate compensatory measures. The CAP would further establish appropriate procedural and process controls to ensure performance of any required compensatory measures.

The NRC staff found that the licensee adequately addressed necessary testing and calibration of the primary and backup SFP level instrument channels to maintain the instrument channels at the design accuracy. Additionally, compensatory actions for instrument channel(s) out-of-service appear to be consistent with guidance in NEI 12-02.

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

4.4 Conclusions for Order EA-12-051

In its letters dated November 18, 2014 [Reference 49], and December 07, 2015 [Reference 18], the licensee stated that they would meet the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed, by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee has conformed to the guidance in NEI 12-02, as endorsed by JLD-ISG-2012-03. In addition, the NRC staff concludes that if the SFP level instrumentation is installed at MNS according to the licensee's proposed design, it should adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit in August 2014 [Reference 17]. The licensee reached its final compliance date on October 8, 2015, and has declared that both of the reactors are in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs which if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders.

6.0 REFERENCES

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. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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. ML13063A185)

11. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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 Number EA-12-049)," August 28, 2013 (ADAMS Accession No. ML13254A204)

12. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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 27, 2014 (ADAMS Accession No. ML14073A462)

13. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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 27, 2014 (ADAMS Accession No. ML14253A187)

14. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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. ML15075A017)

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

16. Letter from Jeremy S. Bowen (NRC) to Steven D. Capps (Vice President – McGuire Site), "William B. McGuire Nuclear Station, Units 1 and 2 – Interim Staff Evaluation Relating to Overall Integrated Plan is Response to Order EA-12-049 (Mitigation Strategies)," January 16, 2014 (ADAMS Accession No. ML13338A406)

17. Letter from Jason Page (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire 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," October 9, 2014 (ADAMS Accession No. ML14241A454)

18. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "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 McGuire Nuclear Station," December 7, 2015 (ADAMS Accession No. ML15343A010)

19. 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)

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

21. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Flood Hazard Reevaluation Report, Response to NRC 10 CFR 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," March 12, 2014, (ADAMS Accession Nos. ML14083A415 and ML14083A417)

22. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 10 CFR 50.54(f) 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 20, 2014, (ADAMS Accession No. ML14098A421)

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

24. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Overall Integrated Plans in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Requirements for Reliable Spent Fuel Pool Instrumentation, Order Number EA-12-051," February 28, 2013 (ADAMS Accession No. ML13086A095)

25. Letter from James Kim (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire Nuclear Station, Units 1 and 2 – Request for Additional Information Re: Overall Integrated Plan in Response to Order EA-12-051, "Reliable Spent Fuel Pool Instrumentation," June 13, 2013 (ADAMS Accession No. ML13157A097)

26. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Response to Request for Additional Information Regarding Overall Integrated Plan in Response to Order EA-12-051, "Reliable Spent Fuel Pool Instrumentation" (TAC Nos. MF1062 and MF1063), July 11, 2013 (ADAMS Accession No. ML13197A409)

27. Letter from Jason Paige (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire Nuclear Station, Units 1 and 2 – Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan for Implementation of Order

EA-12-051, Reliable Spent Fuel Pool Instrumentation, October 28, 2013 (ADAMS Accession No. ML13281A791)

28. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "First Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," August 26, 2013 (ADAMS Accession No. ML13242A009)

29. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Second Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Reliable Spent Fuel Pool Instrumentation Order (Order Number EA-12-051)," February 27, 2014 (ADAMS Accession No. ML14073A467)

30. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Third Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)," August 27, 2014 (ADAMS Accession No. ML14253A187)

31. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Fourth Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Reliable Spent Fuel Pool Instrumentation (SFPI) (Order Number EA-12-051)," February 26, 2015 (ADAMS Accession No. ML15075A017)

32. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Fifth Six-Month Status Reports in Response to March 12, 2012, Commission Order Modifying Licenses With Regard To Reliable Spent Fuel Pool Instrumentation (SFPI) (Order Number EA-12-051)," August 26, 2015 (ADAMS Accession No. ML15246A032)

33. Letter from John Boska (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire Nuclear Station, Units 1 and 2 –Report for the Onsite Audit of Areva Regarding Implementation of Reliable Spent Fuel Pool Instrumentation Related to Order EA-12-051, September 15, 2014 (ADAMS Accession No. ML14203A326)

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

35. 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).

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

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

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

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

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

41. Letter from Eric Leeds (NRC) to Joseph Pollock (NEI), Electric Power Research Institute Final Draft Report, "Seismic Evaluation Guidance: Augmented Approach for Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," As An Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations," May 7, 2013 (ADAMS Accession No. ML13106A331)

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

43. Letter from Frankie G. Vega (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire 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 Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, July 20, 2015 (ADAMS Accession No. ML15182A067)

44. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, "Expedited Seismic Evaluation Process (ESEP) Report (CEUS Sites), Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.2 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," December 17, 2014, (ADAMS Accession No. ML15005A085)

45. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, Docket Nos. 50-369 and 50-370, Renewed License Nos. NPF-9 and NPF-17, 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. ML15253A198)

46. Letter from Juan F. Uribe (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire Nuclear Station, Units 1 and 2 – Interim Staff Response to Reevaluated Flood

Hazards Submitted in Response to 10 CFR 50.54(f) Information Request – Flood-Causing Mechanism Reevaluation, September 3, 2015 (ADAMS Accession No. ML15230A070)

47. COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards," dated November 21, 2014 (ADAMS Accession No. ML14309A256)

48. Letter from Eric J. Leeds (NRC) to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status on the Enclosed List, "Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Regarding Seismic Hazard Re-evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," May 9, 2014 (ADAMS Accession No. ML14111A147)

49. Letter, Duke Energy Carolinas, LLC, "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" - McGuire Nuclear Station Unit 1," November 18, 2014 (ADAMS Accession No. ML14335A322)

50. Letter from Jason Paige (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire Nuclear Station, Units 1 and 2 – Correction Letter to Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan for Implementation of Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," November 7, 2013 (ADAMS Accession No. ML13309A782)

51. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station (MNS), Units 1 and 2, "Expedited Seismic Evaluation Process (ESEP) Closeout, Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," February 4, 2016 (ADAMS Accession No. ML16041A173)

52. COMSECY-15-0019, "Closure Plan for the Reevaluation of Flooding Hazards for Operating Nuclear Power Plants," June 30, 2015, (ADAMS Accession No. ML15153A104)

53. Letter from Nicholas DiFrancesco (NRC) to Steven D. Capps (Vice President – McGuire Site), "McGuire 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. ML16072A038)

54. Letter, Duke Energy Carolinas, LLC, McGuire Nuclear Station, Units 1 and 2, "Supplement to McGuire Nuclear Station, Units 1 and 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,'" May 5, 2016 (ADAMS Accession No. ML16131A634)

55. Nuclear Energy Institute document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 2, December 31, 2015 (ADAMS Accession No. ML16005A625)

56. 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," Revision 1, January 22, 2012 (ADAMS Accession No. ML15357A163)

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Date: June 20, 2016

These reports were required by the order, and are listed in the attached safety evaluation. 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 letters dated October 28, 2013 (ADAMS Accession No. ML13281A791), and October 9, 2014 (ADAMS Accession No. ML14241A454), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated December 7, 2015 (ADAMS Accession No. ML15343A010), Duke submitted its compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

The enclosed safety evaluation provides the results of the NRC staff's review of Duke's strategies for MNS. The intent of the safety evaluation is to inform Duke on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML14273A444). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact John Boska, Orders Management Branch, McGuire Nuclear Station Project Manager, at 301-415-2901 or at John.Boska@nrc.gov.

Sincerely,

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

Docket Nos.: 50-369 and 50-370
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