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

November 9, 2015

Mr. Lawrence J. Weber
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 (TAC NOS. MF0766, MF0767, MF0761, AND MF0762)

Dear Mr. Weber:

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

By letter dated February 27, 2013 (ADAMS Accession No. ML13101A381), Indiana Michigan Power Company (I&M, the licensee) submitted its OIP for Donald C. Cook Nuclear Plant, Units 1 and 2 (CNP) in response to Order EA-12-049. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated January 24, 2014 (ADAMS Accession No. ML13337A325), and August 13, 2014 (ADAMS Accession No. ML14209A122), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letters both dated June 16, 2015 (ADAMS Accession Nos. ML15169A107 and ML15169A106, respectively), I&M 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 27, 2013 (ADAMS Accession No. ML13071A323), I&M submitted its OIP for CNP in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated November 13, 2013 (ADAMS Accession No. ML13310B499), and August 13, 2014 (ADAMS Accession No. ML14209A122), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated December 16, 2014 (ADAMS Accession No. ML14352A231), I&M 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 I&M's strategies for CNP. The intent of the safety evaluation is to inform I&M on whether or not its integrated plans, if implemented as described, provide a reasonable path for compliance with Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. 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, CNP Project Manager, at 301-415-2901 or at John.Boska@nrc.gov.

Sincerely,

Mandy K Halter

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

Docket Nos.: 50-315 and 50-316

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

TABLE OF CONTENTS

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

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

Enclosure

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 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 27, 2013 [Reference 10], Indiana Michigan Power Company (I&M, the licensee) submitted an Overall Integrated Plan (OIP) for Donald C. Cook Nuclear Plant, Units 1 and 2 (CNP, DC Cook) in response to Order EA-12-049. By letters dated August 26, 2013 [Reference 11], February 27, 2014 [Reference 12], August 27, 2014 [Reference 13], and February 25, 2015 [Reference 14], 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 47]. By letters dated January 24, 2014 [Reference 16], and August 13, 2014 [Reference 17], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By two letters dated June 16, 2015 [References 18 and 19], 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 October 1, 2015, the licensee submitted an updated FIP [Reference 44].

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 loss of normal access to the ultimate heat sink (LUHS). Thus, the ELAP with LUHS is used as a surrogate for a BDBEE. The initial conditions and assumptions for the analyses are stated in NEI 12-06, 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 CNP Units 1 and 2 are 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 RCS letdown paths. Decay heat is removed by steaming from the steam generators (SGs) through the SG power-operated relief valves (PORVs) or SG safety valves, and makeup to the SGs is initially provided by the turbine-driven auxiliary feedwater (TDAFW) pump taking suction from the condensate storage tank (CST). Subsequently, the operators would begin a controlled cooldown and depressurization of the RCS by manually operating the SG PORVs. The SGs are first depressurized in a controlled manner to about 290 pounds per square inch gage (psig) and then maintained at this pressure while the operators borate the RCS and close the accumulator isolation valves using electrical power from FLEX generators. This SG depressurization will reduce RCS temperature and pressure. The licensee plans to complete this cooldown within 10 hours of the start of the event. The reduction in RCS temperature will result in inventory contraction in the RCS, with the result that the pressurizer level is expected to indicate empty for some time. Some leakage from the RCP seals is also expected. If the RCS pressure drops below the safety injection accumulator pressure of about 630 psig before the accumulators are isolated (to prevent the injection of nitrogen later in the event), some injection of borated water from the accumulators will occur.

Subsequently, the licensee plans a further depressurization of the SGs in order to further reduce RCS temperature and pressure. When RCS temperature is below 350 °F, RCS pressure is below 350 psig, and plant systems to operate the residual heat removal (RHR) system have been restored, operators will initiate RCS cooling using the RHR system, and reduce RCS temperature below 200 °F.

The dc bus load stripping will be initiated within the first hour to ensure safety-related battery life is extended to 12 hours. Portable FLEX generators will be used to repower battery chargers or instrumentation prior to battery depletion.

The water supply for the TDAFW pump is initially from the CST. The CST will provide a minimum of 12 hours of RCS decay heat removal, in addition to absorbing the latent heat associated with the planned RCS cooldown. Prior to emptying the CST the operators will align the TDAFW pump suction to the essential service water (ESW) supply pipe, which will be pressurized by the FLEX lift pump taking suction from Lake Michigan. In addition, the lift pump can be aligned to feed FLEX booster pumps (one booster pump for each unit, or one booster pump supplying both units) which will be aligned to supply water to the SGs.

Following dc load stripping and prior to battery depletion, two FLEX portable 500 kilowatt (kW), 600 volt alternating current (Vac) diesel generators (DGs) (one per unit) will be deployed from the FLEX storage building (FSB) and connected to power selected 600 Vac motor control centers (MCCs).

RCS makeup and boron addition will conservatively be initiated within 16 hours of the ELAP/LUHS event to ensure that natural circulation, reactivity control, and boron mixing is maintained in the RCS. Two portable electrically-driven FLEX boric acid pumps (one per unit) will be moved from their stored position in the auxiliary building (AB) to take suction from the boric acid storage tanks (BASTs) and inject into the RCS via the charging pump discharge header or the safety injection discharge header. The FLEX boric acid pumps are powered by a single 250 kW FLEX DG that is sized to operate both pumps simultaneously and is deployed from the FSB.

In addition, a National SAFER Response Center (NSRC) will provide high capacity pumps and large turbine-driven DGs to restore one residual heat removal (RHR) cooling train per unit to cool the cores in the long term. There are two NSRCs in the United States.

To maintain SFP cooling capabilities, the licensee determined that it would take approximately 49 hours for pool water level to drop to a level requiring cooling or the addition of makeup to preclude fuel damage conservatively assuming a dual unit full core offload. Makeup water would be provided using the FLEX lift pump drawing on Lake Michigan and discharging through a hose which will be connected to add water to the SFP. Ventilation of the generated steam is accomplished by opening building rollup doors and the SFP roof fire dampers thus establishing a natural draft vent path. The SFP is located in a section of the AB and serves both units.

For Phases 1 and 2 the licensee's calculations demonstrate no actions are required to maintain containment pressure below design limits for over 72 hours. In Phase 2, the licensee will power 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, containment cooling and depressurization would be accomplished by operating one hydrogen skimmer fan and circulating the air through the ice condenser. The skimmer fans would be powered by a 4160 Vac turbine-driven DG supplied by the NSRC.

Below are specific details on the licensee's strategies to restore or maintain core cooling, SFP and containment 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

In accordance with Order EA-12-049, licensees are required to maintain or restore cooling to the reactor core in the event of an ELAP concurrent with a LUHS. Although the ELAP results in an immediate trip of the reactor, sufficient core cooling must be provided to account for fission product decay and other sources of residual heat. Consistent with endorsed guidance from NEI 12-06, Phase 1 of the licensee's core cooling strategy credits installed equipment (other than that presumed lost to the ELAP/LUHS) 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/LUHS 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/LUHS event. Maintenance of sufficient RCS inventory, despite ongoing system leakage expected under ELAP conditions, is accomplished through a combination of installed systems and FLEX equipment. The specific means used by the licensee to accomplish adequate core cooling during the ELAP/LUHS 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

The FIP states that in an ELAP event operators would initiate RCS cooldown by depressurizing the SGs at the maximum allowable rate until a SG pressure of about 290 psig is reached. Core cooling would be accomplished by natural circulation flow in the RCS using the SGs as the heat sink. SG inventory makeup would be promptly initiated using the TDAFW pump taking suction from the CST, with steam vented from the SGs via the PORVs. The TDAFW pump SG injection motor operated valves (MOVs) are powered from the N Train battery; however, there are operating procedures that provide guidance for operating these valves locally if the battery is depleted. Local manual SG PORV operation is credited because there are components in the control systems which are not located in robust structures. In light of credit for the recently installed SHIELD® low leakage RCP seals, initiation of RCS cooldown is planned within 8 hours of the ELAP. Completion of the cooldown is projected to occur within the following 2 hours. The FIP states that 215,226 gallons of feedwater per unit to the SGs is sufficient to support post trip RCS cooldown and decay heat removal for at least 12 hours. The CNP Technical Specifications require each CST to have at least 182,000 gallons, and the licensee stated in the FIP that each unit's CST is normally maintained at about 405,000 gallons during power operation.

All four SGs in each unit will be used to maintain a symmetric RCS cooldown for the first 24 hours. Thereafter, with the RCS having been fully borated and well mixed, the licensee states that RCS cooling may be accomplished by two of four SGs to reduce the operator workload required to manually operate the SG PORVs.

3.2.1.1.2 Phase 2

Upon depletion of the CST inventory, Lake Michigan, the UHS, would be used as the credited SG cooling water source. A portable diesel-driven FLEX Lift Pump would be deployed to take suction from Lake Michigan at the circulating water forebay using a suction hose and strainer. The FLEX Lift Pump would discharge via a hose connection to a section of ESW piping, which has a connection to the TDAFW suction. The TDAFW discharge MOVs are powered from the N Train battery; thus, during Phase 2, power to the N Train battery charger would be provided by a 600 Vac, 500 kW FLEX diesel generator or a 480 Vac, 350 kW diesel generator.

The alternate strategy for supplying water to the SGs involves routing the discharge of the FLEX Lift Pump to a FLEX Booster Pump to achieve sufficient pressure to feed the SGs following the initial plant cooldown to 290 psig in the SGs. This strategy involves the use of multiple manifolds to route the FLEX Booster Pump discharges to all four SGs by using connection points on the west motor driven auxiliary feedwater pump discharge header and existing drain connections on the main feedwater header. The FIP also indicated that the SGs in RCS loops 2 and 3 could be supplied with cooling water through a cross-tie from a FLEX Booster Pump connected to the opposite unit's west motor driven auxiliary feedwater pump discharge header.

3.2.1.1.3 Phase 3

Additional equipment from an NRSC is scheduled to arrive 24 hours after it is requested by the licensee. The FIP states that two 1 megawatt (MW) generators would provide 4 kV power to Train B components on one unit, which would allow operation of equipment necessary to establish RHR cooling using the west component cooling water (CCW) pump and the west RHR pump on that unit. The NSRC will provide four 1 MW generators to provide the necessary power to both units. In addition, the NSRC would also provide, per unit, a diesel-driven, low pressure-high flow raw water pump and two hydraulically driven floating lift pumps with a diesel driven hydraulic driver unit to provide suction flow to the raw water pump.

The licensee stated that the two NSRC raw water pumps would provide flow to the ESW systems through two connections to the ESW pump discharge strainers, which is accomplished with two replacement FLEX strainer lids equipped with hose connections to accept discharge from the NSRC raw water pumps.

Each NSRC raw water pump would take suction from the circulating water forebay using two hydraulically driven floating lift pumps and discharge through the selected pump discharge strainer to the associated ESW pump discharge header. This would restore ESW system cooling flow to one CCW heat exchanger and the control room ventilation systems in each unit. This would allow core cooling to be accomplished using RHR cooling.

After the initial depressurization of the SGs to 290 psig, the licensee plans a further depressurization of the SGs in order to further reduce RCS temperature and pressure and allow the RHR system to be placed in operation for core cooling. While the SGs are at 290 psig the operators will borate the RCS and close the accumulator isolation valves using electrical power from FLEX generators. Subsequently, operators will initiate a further depressurization of the SGs in order to further reduce RCS temperature and pressure. When RCS temperature is below 350 °F, RCS pressure is below 350 psig, and plant systems to operate the residual heat removal (RHR) system have been restored, operators will initiate RCS cooling using the RHR system, and reduce RCS temperature below 200 °F. The licensee plans to use the reactor vessel head vent valves to reduce the RCS pressure below 350 psig if necessary.

3.2.1.2 RCS Makeup Strategy

3.2.1.2.1 Phase 1

As stated in the FIP, no actions would be needed to maintain adequate RCS inventory in Phase 1. The licensee has installed Generation 3 SHIELD® low leakage RCP seals. These low leakage seals limit the seal leakage rate to 1 gallon per minute (gpm) per RCP. Further considering 1 gpm of additional RCS leakage results in an assumed total RCS leak rate of no more than 5 gpm. With credit for the SHIELD® seals and the passive injection of accumulator inventory, Westinghouse analyses demonstrate that natural circulation in the RCS could be maintained for multiple days under postulated ELAP conditions without reliance upon FLEX RCS injection. However, the licensee conservatively determined that additional RCS boration should be initiated by 16 hours into the event to ensure adequate shutdown margin. This RCS boration is discussed in the Phase 2 section below.

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

3.2.1.2.2 Phase 2

The FIP states that during the cooldown described, borated water would be added to compensate for the positive reactivity of the cooldown and xenon decay. Makeup to the RCS would also compensate for inventory contraction caused by the RCS cooldown and the small amount of RCS leakage. Two portable electric-powered FLEX boric acid pumps (one per unit) would be used to inject boric acid into the RCS. The FLEX boric acid pumps are powered from a single FLEX 250 kW diesel generator, which is deployed from the FSB.

The licensee stated that RCS boration would be initiated within 16 hours and be completed within 24 hours of the ELAP to ensure a symmetric RCS cooldown and adequate boric acid mixing via natural circulation flow. Active SG cooling and natural circulation of all four RCS loops is maintained for at least one hour following boron injection to ensure boric acid mixing for long-term core subcriticality considering a cooldown to an RCS temperature below 200 °F. The licensee's analysis demonstrates subcriticality down to 201 °F (Mode 4) with the most reactive rod stuck out. This ELAP event does not require the assumption of a stuck rod, and that additional negative reactivity would allow an additional temperature reduction that the licensee has not quantified.

The portable FLEX boric acid pumps would take suction from the boric acid storage tanks (BASTs), from a hose connected to the boric acid transfer pump suction header, and discharge to the RCS through a tee connection installed on the chemical and volume control system (CVCS) charging pumps discharge header. The RCS injection flow path is through the boron injection tank into the RCS through the four RCS cold legs. An alternate connection for RCS make up would be available through vent and drain connections on the safety injection system (SIS) pump discharge piping by connecting a portable SIS manifold to the discharge of the FLEX boric acid pump. The licensee stated that one BAST contains sufficient volume to maintain core subcriticality in one reactor following the cooldown. The CNP Updated Final Safety Analysis Report (UFSAR) Section 9.2.2 states that there are three BASTs that are shared by Units 1 and 2. Following depletion of the BASTs, the refueling water storage tanks (RWSTs) will be used.

Per the FIP, the design discharge pressure for the FLEX boric acid pumps is 1550 psig. There is a possibility in some ELAP scenarios that the RCS pressure could exceed 1550 psig or that the RCS could approach water-solid conditions. In particular, such conditions may come about due to the injected coolant compressing or completely collapsing the vapor bubbles that would be expected to form in the pressurizer and reactor vessel upper head following RCS depressurization. In this case, venting of the RCS would be necessary to support the injection of borated makeup. Using procedure FSG-8, "Alternate RCS Boration," the licensee will preferentially vent using the reactor vessel head vent system. The PORVs on the pressurizer would be available as an alternative if the head vent system is not available. The staff notes that, consistent with NUREG-0737, "Clarification of TMI Action Plan Requirements," the reactor vessel head vent system was designed to provide reliable capability to vent noncondensable gas or steam from the reactor vessel head under post-accident conditions. The NRC staff notes that use of the reactor vessel head vent system is preferable to pressurizer PORVs for two main reasons: (1) use of the smallest vent path capable of providing the required letdown is desirable, especially under ELAP/LUHS conditions where the availability of high pressure pumps and borated makeup may be limited, and (2) the reactor vessel head vent system is safety-related and has two flow paths, each with redundant isolation valves, which provides increased confidence in the capability to isolate the vent path when it is no longer required. Therefore, the NRC staff agrees with the preferential use of the head vent system.

3.2.1.2.3 Phase 3

In its updated FIP, the licensee stated that RCS inventory control strategy in Phase 3 would use the same inventory control methods as described for Phase 2, which involves the use of a FLEX boric acid pump powered by a FLEX 250 kW diesel generator.

The licensee indicated that initially available clean water sources will be used such as the RWST and if needed Lake Michigan water is available for indefinite use. Due to (1) the large quantity of reactor-grade coolant available in the RWSTs, (2) the installation of low-leakage seals, and (3) the substantial reduction in RCS long-term leakage rate expected after the reactor has been depressurized and cooled to approximately 200 °F, the staff does not consider it necessary for the licensee's pre-planned mitigating strategy to provide additional sources of purified water for RCS makeup. However, in the long-term recovery phase, water quality and the need for treating the water used for makeup would need to be addressed.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

In the FIP, section 2.10.2, the license states that the current design-basis flood will remain below the current lakeside seawall level, and that flooding of the plant site would not occur. Therefore, there are no variations to the core cooling strategy in the event of a flood. Refer to section 3.5.2 of this safety evaluation (SE) for further discussion on flooding.

3.2.3 Staff Evaluations

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

In NEI 12-06 the guidance states 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.

3.2.3.1.1 Plant SSCs

The licensee's Phase1 core cooling FLEX strategy relies on the TDAFW pump for each unit to provide AFW flow to a common header that feeds all four SGs. In UFSAR Section 10.5.2.3 it states that the TDAFW pump is housed in a missile-protected enclosure. Furthermore, UFSAR Section 2.9.4 states that the auxiliary feed pumps are Seismic Class I equipment supported by the foundation slab designed to Class I criteria within the Turbine Building. Seismic Class I equipment is designed to remain functional following a design basis earthquake. The NRC staff noted that the TDAFW pumps are located in a temperature-controlled area of the Turbine Building. In UFSAR Section 2.8.7 it states that plant grade and the design bases of features related to plant safety are established to consider the coincidence of the maximum seiche postulated for the site with the highest recorded lake level; thus, the TDAFW system is flood-protected. The staff finds that the TDAFW pumps are robust and are expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3. Equipment operation during an ELAP event will be addressed in Section 3.9.1 of this SE.

The licensee's Phase 1 core cooling FLEX strategy relies on the SG PORVs to vent steam from the SGs for a controlled cooldown. In UFSAR Section 10.2.2 it states that a power relief valve and a bank of five safety valves are installed on each main steam line after it exits the containment and downstream of the safety valves an SG stop valve is installed in each line as close to the containment wall as possible. Furthermore, UFSAR Section 10.2.3 states that the main steam system up to and including the SG stop valves is designed to Seismic Class I criteria. In its FIP the licensee stated that the SG PORVs are located in the Seismic Category I Auxiliary Building and the staff noted that UFSAR Section 2.9.5 states the concrete walls and roof of the Auxiliary Building were designed to withstand the design-basis tornado missiles. In UFSAR Section 2.8.7 it states that plant grade and the design bases of features related to plant safety are established to consider the coincidence of the maximum seiche postulated for the site with the highest recorded lake level; thus, the SG PORVs are flood-protected. The staff finds that the SG PORVs are robust and are expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

The licensee's Phase 1 core cooling FLEX strategy relies on its CSTs as the water source for the TDAFW pumps. The staff's evaluation of the robustness and availability of the CSTs for an ELAP event is discussed in SE Section 3.10.1.

The licensee's Phase 2 core cooling FLEX strategy relies on the continued use of its TDAFW pump with its suction from a FLEX Lift pump or the use of a FLEX Lift pump and a FLEX Booster pump. The staff's evaluation of the robustness and availability of FLEX connection points for the FLEX Lift pump and FLEX Booster pump is discussed in SE Section 3.7.3.1.

The licensee's Phase 2 core cooling FLEX strategy relies on Lake Michigan upon depletion of the CST inventory to be the credited SG cooling water source. The staff's evaluation of the robustness and availability of Lake Michigan for an ELAP event is discussed in SE Section 3.10.2.

The licensee's Phase 3 core cooling FLEX strategy relies on the use of the west CCW pump, west RHR pump and flow to the ESW system on each unit via a connection to the west ESW pump discharge strainer on each unit. In UFSAR 2.9.2 it specifies that the CCW System, ESW System, and RHR System are Seismic Class I components. The FIP states that the ESW System includes two duplex strainers located in the Seismic Class I ESW Screenhouse. The licensee's strategy relies on the west ESW pump discharge strainer cover being removed and replaced with a FLEX strainer lid equipped with hose connections to accept the discharge from the NSRC raw water pump. The FLEX strainer lid adapters and hose manifolds are fabricated for each unit and are stored in the FSB. The staff finds that the ESW system is robust and available during an ELAP event consistent with NEI 12-06, Section 3.2.1.3. In addition, since the FLEX strainer lid adapters are protected in the FSB they are also considered available during an ELAP event.

The licensee's Phase 1 RCS inventory control FLEX strategy relies on the Generation 3 SHIELD® low leakage RCP seals and Westinghouse analyses have demonstrated that no FLEX RCS make up is needed prior to 16 hours; thus, the licensee's strategy does not rely upon any other plant SSCs.

The licensee's Phase 2 and Phase 3 RCS inventory control FLEX strategies rely on the use of a portable electric FLEX boric acid pump for each unit, with one FLEX 250 kW DG powering the two pumps, to inject boric acid into the RCS. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX boric acid pump is discussed in SE Section 3.7.3.1. The FIP stated that the three FLEX boric acid pumps (one for each unit and a spare) are stored in the Auxiliary Building and the FLEX 250 kW DG (and a spare) are stored in the FSB. The staff finds that the FLEX boric acid pumps and FLEX 250 kW diesel generators are protected from applicable external hazards and would be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

The licensee's Phase 2 RCS inventory control FLEX strategy relies on the use of the BASTs as the borated water source. The FIP states that one BAST contains sufficient volume to maintain core subcriticality in one reactor following the cooldown. The staff's evaluation of the robustness and availability of the BASTs for an ELAP event is discussed in SE Section 3.10.2.

The licensee indicated that if needed Lake Michigan water is available for indefinite use for injection into the RCS. The staff's evaluation of the robustness and availability of the UHS, Lake Michigan, for an ELAP event is discussed in SE Section 3.10.2.

3.2.3.1.2 Plant Instrumentation

According to the licensee's FIP, control room instrumentation would be available due to the 12 hour 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 key credited plant parameters would be available, as stated below.

- An operator would be dispatched to obtain local CST level indication (in the TDAFW pump room)

- At the control room, control racks, and reactor cable tunnel quad 3, using procedure FSG-711 and electronic multimeters, instrument readings are available for the following:

    - SG Wide Range Level (all four generators)
    - Pressurizer level
    - SG Pressure (all four loops)
    - RCS wide range pressure (loops 1 and 2)
    - RCS wide range temperature (loops 1 and 3)
    - Lower Containment Pressure (all four quadrants)
    - Incore temperatures (5 locations)

- In the inverter room, the Train B DG room, the Train B Reactor Vessel Level Instrumentation System (RVLIS) cabinet, and the Channel 1 Gamma-Metrics Neutron Flux Monitor cabinet, via use of temporary power from portable generators:

    - Source Range Nuclear Instrumentation
    - Narrow/Wide Range Reactor Vessel Level
    - Reactor Vessel Upper Plenum Level
    - RCS Wide Range Pressure
    - Wide Range Log Power
    - Wide Range Startup rate
    - RCS Loop 1 wide range temperature – hot leg/cold leg

- At the containment penetration, by using procedure FSG-712 to connect portable equipment:

    - SG Narrow Range Level (all 4)
    - Pressurizer level
    - RCS loop 2 wide range pressure
    - RCS Loop 4 narrow range temperature (hot leg and cold leg)
    - Incore temperatures (5 locations)
    - Reactor vessel narrow range level
    - Gamma-metrics source and wide range nuclear instrumentation

- At the containment penetration, via local or test gauges:
    - SG Pressure (all 4)
    - Containment pressure (lower)

The licensee's FIP states that, as recommended by Section 5.3.3 of NEI 12-06, procedures have been developed to read the above instrumentation locally using a portable instrument, where applicable.

3.2.3.2 Thermal-Hydraulic Analyses

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

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

Accordingly, the ELAP coping time prior to providing makeup to the RCS is limited to the duration over which the flow in the RCS remains in natural circulation, prior to the point where continued inventory loss results in a transition to the reflux or boiler-condenser cooling mode. In particular, for PWRs with inverted U-tube SGs, the reflux cooling mode is said to exist when vapor boiled off from the reactor core flows out the saturated, stratified hot leg and condenses on SG tubes, with the majority of the condensate subsequently draining back into the reactor vessel in countercurrent fashion. Quantitatively, as reflected in documents such as PWROG-14064-P [Reference 49], 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. 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 [Reference 47]. Subsequently, a series of additional simulations performed by the staff and vendor 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.

The licensee produced a plant-specific analysis, Westinghouse calculation note CN-FSE-13-13-R, "D.C. Cook Unit 1 and Unit 2 (AEP/AMP) Reactor Coolant System (RCS) Inventory Control and Long-Term Subcriticality Analysis to Support the Diverse and Flexible Coping Strategy (FLEX)," Rev. 1, dated October 15, 2014, which presents analyses showing that, with credit for the SHIELD® low leakage seals, no FLEX borated make up would be required to ensure adequate shutdown margin or RCS inventory prior to 16 hours. Because of its credit for SHIELD® low leakage RCP seals, the licensee has extended the allowable time to initiate an RCS cooldown to eight hours while still targeting completion of the initial cooldown to 290 psig in the SGs within the next two hours. Based on this calculation, the licensee concludes that sufficient margin to avoid reflux cooling is available. The NRC staff's review of the plant-specific analysis in CN-FSE-13-13-R determined that simplified and approximate calculation methods had been used. In light of (1) the licensee's installation of SHIELD® seals, which should extend the duration over which natural circulation flow can be maintained in the RCS well beyond 16 hours and (2) the staff's review of more-detailed thermal-hydraulic calculations for the ELAP event for a variety of assumed RCS leakage rates, the staff concluded that the licensee's strategy for RCS makeup provides sufficient margin to the onset of reflux cooling.

Therefore, based on the evaluation above, which demonstrates large margins, the NRC staff believes that the licensee's analysis is acceptable for determining the sequence of events, including time-sensitive operator actions, and 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 the 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 increased leakage and the potential for failure of elastomeric o-rings and other components, which could further increase the leakage rate. As discussed above, as long as adequate inventory is maintained in the RCS, natural circulation can effectively transfer residual heat from the reactor core to the SGs and limit local variations in boric acid concentration. Along with cooldown-induced contraction 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.

Per the FIP, the licensee credits Generation 3 SHIELD® low leakage seals for FLEX strategies including RCS inventory control and boration. The low leakage seals limit the total RCS leak rate to no more than 5 gpm (1 gpm per RCP seal and 1 gpm of unidentified RCS leakage).

The SHIELD® low leakage seals are credited in the FLEX strategies in accordance with the four conditions identified in the NRC's endorsement letter of TR-FSE-14-1-P, "Use of Westinghouse SHIELD Passive Shutdown Seal for FLEX Strategies" dated May 28, 2014 (ADAMS Accession No. ML14132A128). In its FIP, the licensee describes compliance with each condition of SHIELD seal use as follows:

(1) Credit for the SHIELD® seals is only endorsed for Westinghouse RCP Models 93, 93A, and 93A-1.

    CNP, Unit 1 and Unit 2 compliance letter: The CNP Unit 1 and Unit 2 RCPs are Model 93AS. The "S" designation refers to the presence of a spool piece between the pump and the motor that facilitates seal inspection and replacement. The seal package for Model 93A RCPs is identical to that for Model 93AS.

(2) The maximum steady-state reactor coolant system (RCS) cold-leg temperature is limited to 571 °F during the ELAP (i.e., the applicable main steam safety valve setpoints result in an RCS cold-leg temperature of 571 °F or less after a brief post-trip transient).

    CNP, Unit 1 and Unit 2 compliance letter: The maximum steady-state RCP seal temperature during an ELAP response is expected to be the RCS cold leg temperature corresponding to the lowest SG safety relief valve setting of 1065 pounds per square inch gage (psig). This corresponds to an RCS cold leg temperature of approximately 557 °F.

(3) The maximum RCS pressure during the ELAP (notwithstanding the brief pressure transient directly following the reactor trip comparable to that predicted in the applicable analysis case from WCAP-17601-P) is as follows: For Westinghouse Models 93 and 93A-1 RCPs, RCS pressure is limited to 2250 psia; for Westinghouse Model 93A RCPs, RCS pressure is to remain bounded by Figure 7.1-2 of TR-FSE-14-1-P, Revision 1.

CNP, Unit 1 and Unit 2 compliance letter: Normal Unit 1 and Unit 2 operating pressures are 2085 psig and 2235 psig, respectively. Assuming a plant cooldown is initiated at the maximum allowed time of 8 hours following the ELAP and the cooldown and depressurization is completed within 2 hours, the licensee expects that the plant pressure would remain bounded by Figure 7.1-2 of TR-FSE-14-1-P, Revision 1, which shows a limit of 2250 psig for the first 24 hours.

(4) Nuclear power plants that credit the SHIELD® seal in an ELAP analysis shall assume the normal seal leakage rate before SHIELD® seal actuation, and a constant seal leakage rate of 1.0 gallon per minute for the leakage after SHIELD® seal actuation.

CNP, Unit 1 and Unit 2 compliance letter: A constant Westinghouse SHIELD® RCP seal package leak rate of 1 gpm per RCP was assumed in the applicable analysis, CN-FSE-13-13-R. Assumption of the normal seal leakage rate until SHIELD® seal actuation occurred would result in a small volume of additional leakage that would have an inconsequential effect on the analysis results.

During the ELAP event, even after the actuation of the SHIELD® seal, several o-rings inside the RCP may be exposed to elevated pressure and temperature conditions. The specific o-rings that would be affected depend on the particular RCP model. The NRC staff discussed the issue with the licensee during the audit. The licensee stated that, in the future, only high-temperature-qualified o-rings would be installed in locations where the potential exists for exposure to elevated pressure and temperature conditions.

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
    - initially increases above its equilibrium value following reactor trip, thereby adding negative reactivity
    - peaks at roughly 12 hours and subsequently decays away gradually, thereby adding positive reactivity

- the injection of borated makeup from passive accumulators due to the depressurization of the RCS, which adds negative reactivity

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

The NRC staff requested that the industry provide additional information to justify that borated makeup would adequately mix with the RCS volume under natural circulation conditions potentially involving two-phase flow. In response, the Pressurized Water Reactor Owner's Group (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. In a letter dated January 8, 2014 [Reference 48], 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.

According to Westinghouse letter LTR-FSE-13-66, "Response to NRC Audit Question 16 Regarding the FLEX Integrated Plan Submittal for D.C. Cook Units 1 and 2," dated November 1, 2013 (proprietary), the licensee confirmed it is complying with the August 15, 2013, position paper on boric acid mixing. The letter does not state compliance with conditions of the NRC endorsement letter, however the methodology used by CNP addresses the NRC stated conditions.

According to the FIP, RCS boration will be initiated no later than 16 hours following an ELAP event and completed within 24 hours. The quantity of boric acid the licensee plans to add to the RCS is intended to provide adequate shutdown margin for a xenon-free condition at an RCS temperature of approximately 201 °F. The licensee considers the planned completion time for this action to be conservative, as significant negative reactivity due to xenon remains well past 24 hours following ELAP initiation. Active SG cooling and natural circulation of all four loops is maintained for at least one hour following boron injection to ensure boron mixing for long-term core subcriticality. Based on the information presented during the audit, the staff could not specifically confirm whether RCS boration would be completed within 24 hours. Nevertheless, the staff concluded that the licensee's mitigating strategy would provide adequate shutdown margin because (1) the extended cooldown to approximately 200 °F would occur after the completion of RCS boration and (2) significant negative reactivity from xenon would exist well beyond 24 hours.

The BASTs would be the primary suction source of borated water for FLEX RCS makeup. There are three BASTs, each with a capacity of 11,000 gallons. Per the CNP Technical Requirements Manual, TRM 8.1.1, in Modes 1 and 2 one BAST must be operable for each unit, with at least 8,500 gallons of water with ≥ 6,550 parts per million (ppm) boron concentration in an operable BAST. The FIP states that one BAST has sufficient volume to maintain one reactor core subcritical following an RCS cooldown to Mode 4 (201 °F). Per the FIP, Phase 3 RCS boration and inventory control would use the same methods as described in Phase 2. In the OIP, the licensee stated that the RWSTs were an alternate source of borated water.

In LTR-FSE-13-66 it states that the shutdown margin calculation performed for the ELAP event was based on existing cycle-specific shutdown margin calculations. In NEI 12-06, section 11.8.2, it states that 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 constitute changes to the plant design, the staff expects that any changes to the core design, such as a core reload analysis, will be evaluated to determine that they do not adversely impact the approved FLEX strategies, especially the analyses which demonstrate that no recriticality will occur during a FLEX RCS cooldown.

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 Equipment and Water Supplies

The licensee's Phase 2 core cooling FLEX strategies rely on the use of a FLEX Lift pump and/or FLEX Booster pump to support injection to the SGs. A single trailer-mounted FLEX Lift pump, which is a diesel-driven centrifugal pump, would be deployed to draw suction from the UHS from the circulating water forebay and discharge to the suction of each unit's TDAFW pump using the ESW piping connection to the TDAFW pump. The design pressure of the ESW system is 105 psi and the FLEX Lift pump injection pressure into the ESW system would be controlled to ensure the ESW system design pressure is not exceeded.

In addition, the discharge of the FLEX Lift pump can be routed to a FLEX Booster pump to achieve sufficient pressure to feed all the SGs in both units. The FLEX Booster pumps are also diesel-driven centrifugal pumps that are trailer-mounted and stored with an associated hose trailer. One FLEX Booster pump can raise the water delivery pressure to the SGs to at least 327 gpm at 300 psia at the SG feed ring.

During the audit, the licensee provided calculation MD-12-FLEX-002-S, "DC Cook FLEX Core Cooling and SFP Makeup Hydraulic Analysis," which determined the necessary pump performance criteria for the portable FLEX pumps (Lift pump and Booster pump) to support the licensee's core cooling strategies. The staff noted that this calculation assessed numerous possible lineups based on such variables as suction sources, connection points and hose paths to determine adequate performance criteria for the FLEX Lift pump providing suction for the TDAFW pump in both units and for the FLEX Lift and FLEX Booster pumps providing injection into the SGs in both units.

During the audit, the NRC staff performed a walkdown of the licensee's core cooling FLEX strategies and noted that the point of deployment for the portable FLEX pumps, hose routing and deployment connection points (primary and alternate) were consistent with the licensee's hydraulic analysis. The staff noted that the capability of the FLEX Lift pump and FLEX Booster pump are identified in their respective procedures, 12-OHP-4027-FSG-311, "FLEX Lift Pump Operation," and 12-OHP-4027-FSG-312, "FLEX Booster Pump Operation." Operators will be present as necessary while the FLEX Lift pump and FLEX Booster pump are in operation in order to control and maintain proper flow to support SG injection based on information from the control room.

A single FLEX Lift pump can support both Units 1 and 2 for core cooling, or provide SFP cooling, by lifting water from the forebay (which is connected to Lake Michigan by the North and South intake tunnels) and supplying water to an AFW pump, or to a FLEX Booster pump, or directly to the SFP. The NRC staff noted that the procedure for the FLEX Booster pump indicates that it is capable of boosting pressure to at least 400 psig at a flow rate of up to 600 gpm (75 gpm to each SG in both units) with a suction from the FLEX Lift pump. In its updated FIP, the licensee stated that the combination of one FLEX Lift pump and one FLEX Booster pump is capable of providing adequate flow to two SGs in Unit 1 and two SGs in Unit 2 concurrently, which can remove all the core decay heat.

The licensee's Phase 3 core cooling FLEX strategies rely on the following pumps provided by the National SAFER Response Center (NSRC): a diesel-driven, low pressure, high flow, raw water pump (5000 gpm and 150 psi) to provide flow to the ESW system, and two hydraulically driven, floating lift pumps with a diesel driven hydraulic driver unit (26 feet water lift and 5000 gpm) to provide flow to the raw water pump. These pumps from the NSRC would be used in conjunction with the west CCW pump and west RHR pump to establish RHR cooling. During the audit, the staff noted the licensee's procedure, 1/2-OHP-4027-FSG-13, "Alternate RHR Cooling," provides guidance for placing the necessary portions of the ESW system, CCW system, and RHR system in service for decay heat removal. In addition, the licensee's procedure, 1/2-OHP-4027-FSG-1301, "Alternate RHR Cooling Equipment Deployment," provides guidance for deploying the NSRC raw water pump, the NSRC floating lift pumps and their hydraulic driver unit, and the associated ESW strainer lid adapter, hoses, manifolds, and fittings.

The licensee's Phase 2 RCS inventory control strategies rely on the use of a FLEX Boric Acid Pump with suction from the BASTs to support injection into the RCS through the charging pumps discharge header or through the safety injection pump discharge piping. The FIP stated that there are three FLEX Boric Acid Pumps (one for each unit and a spare) that are positive displacement pumps rated for 26 gpm at 1550 psig and powered by a FLEX 250 kW diesel generator.

During the audit, the licensee provided calculation MD-12-FLEX-001-S, "DC Cook FLEX – RCS Makeup Hydraulic Analysis," which determined the necessary pump performance criteria for the portable FLEX Boric Acid pump to support the licensee's RCS inventory control strategies. The staff noted that this calculation assessed numerous possible lineups based on such variables as connection points and hose paths to determine adequate performance criteria for the FLEX Boric Acid pump taking suction from the BASTs.

During the audit, the NRC staff performed a walkdown of the licensee's RCS inventory control FLEX strategies and noted that the point of deployment for the portable FLEX pumps, hose routing and deployment connection points (primary and alternate) were consistent with the licensee's hydraulic analysis. The staff noted that the capability of the FLEX Boric Acid Pump is identified in procedure, 12-OHP-4027-FSG-811, "FLEX Boric Acid Operation." This procedure also indicates that intermittent use of the pump should be considered to limit heat up of the pump components. Operators will be present as necessary while the FLEX Boric Acid pump is in operation in order to control and maintain proper flow to support RCS inventory control based on information from the control room.

Based on its review, the NRC staff concludes that, if implementation is performed as described, the licensee has demonstrated that its FLEX portable pumps are capable of supporting the water make-up to the SGs and RCS and of drawing suction from Lake Michigan to support the FLEX strategies.

3.2.3.6 Electrical Analyses

The CNP electrical FLEX strategies are identical 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. Furthermore, the electrical coping strategies are the same for all modes of operation.

According to the CNP FIP, ELAP entry conditions can be verified by control room staff. An ELAP would be declared after CNP validates that offsite power and the emergency DGs (EDGs) are not available. This step is time sensitive and needs to occur within 15 minutes following the start of the event. During the first phase of the ELAP event, CNP will be relying on the safety-related Class 1E station batteries to cope until additional power supplies (i.e., FLEX DGs) can be aligned and connected to the CNP electrical distribution system (Phase 2). Transitioning to Phase 2 includes aligning and placing into service 600 Vac (500kW) and 480 Vac (350 kW, 250 kW, 26 kW) FLEX DGs. The 600 Vac FLEX DGs (the primary strategy) would provide power to vital battery chargers, battery room exhaust fans, one boric acid transfer pump, the middle boric acid evaporator feed pump, Train B hydrogen igniters, and Train A RVLIS. The 480 Vac, 350 kW FLEX DG (the alternate strategy) would re-power the existing 480/600 V Outside Temporary Outage Power Transformer. This FLEX DG would also provide an alternate capability to provide power to the hydrogen igniters, RVLIS, SI Accumulator outlet valves, the N Train battery charger for both units, and the Unit 1 N Train battery room exhaust fan. The licensee stated in its compliance letter that the Unit 2 N Train battery room exhaust fan could be powered from the Phase 3 DGs, and that a calculation shows that with no ventilation it would take at least 65 hours for the hydrogen concentration in the Unit 2 N Train battery room to reach 2% (which is below the flammable limit), providing time to align the Phase 3 DGs. The 250 kW FLEX DG would power the FLEX Boric Acid Pump Electric Motor. The 26 kW FLEX DGs are available to power portable ventilation for the control room, TDAFW pump room, and the Train A and Train B battery rooms.

If the 250 Vdc Vital Chargers are not energized and thus not supplying the 250 Vdc Vital Batteries, then CNP's plan directs operators to complete a dc deep load shed for any Vital Battery not being supplied by a battery charger within 60 minutes following the start of the ELAP event. This would ensure that the 250 Vdc Vital Batteries could supply power for a 12-hour coping duration and provide sufficient time to align and connect the FLEX DGs to the CNP electrical distribution system.

The licensee verified the separation and isolation of the FLEX DGs from the Class 1E EDGs, and the capacity of the FLEX DGs, through the following calculations and documents:

- 1-E-S-600V-FLEX-001, "500 kW N Strategy FLEX Event Diesel Generator Analysis," Rev. 0
- 2-E-S-600V-FLEX-001, "500 kW N Strategy FLEX Event Phase 2 Diesel Generator Analysis," Rev. 0
- 1-E-S-600V-FLEX-002, Rev. 0, "Diesel Generator and Cable Sizing and Ampacity for FLEX Phase 2 Strategies"
- 2-E-S-600V-FLEX-002, "Diesel Generator and Cable Sizing and Ampacity for FLEX Phase 2 Strategies", Rev. 0
- 12-E-S-480-FLEX-001, "Boric Acid FLEX Pump Electrical Analysis," Rev. 0
- FLEX DG manufacturer specification sheets
- Conceptual single line electrical diagrams
- procedures that direct operators how to align, connect, and protect associated systems and components

The NRC staff review confirmed that the FLEX DGs have sufficient capacity and capability to supply the necessary loads during an ELAP event.

During the audit, the licensee provided dc system analysis, calculation 12-E-S-250D-FLEX-001, "250VDC Battery Deep Load Shed (DLS) Analysis," Rev. 0, which verified the capability of the dc system to supply the required loads during the first phase of the CNP FLEX mitigation strategy plan for an ELAP event. The licensee's analysis identified the required loads and their associated ratings (amperage and minimum voltage) and loads that would be shed to ensure battery operation for at least 12 hours. The licensee expects that power will be restored to the battery charger within 12 hours. The licensee stated it had followed NEI white paper, EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern, (ADAMS Accession No. ML13241A186), which was endorsed by NRC (ADAMS Accession No. ML13241A188).

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 (ADAMS Accession No. ML15148A418). 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 IEEE Standard 485-2010, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," or using an empirical algorithm described in the report.

Based on the information contained in NUREG/CR-7188, and the staff's review of the licensee's analysis, the battery vendor's capacity and discharge rates for the batteries, the guidance in CNP procedures 1-OHP-4027-FSG-4, "ELAP Power Management," Rev. 0, and 2-OHP-4027-FSG-4, "ELAP Power Management," Rev. 0, the NRC staff found that CNP's load shed strategy is acceptable and that the batteries are expected to have sufficient capacity to supply power to required loads for at least 12 hours.

For Phase 3, CNP plans to implement core cooling with the RHR system using electrical power from offsite equipment/resources. The offsite resources that will be provided by the NSRCs include two 1-MW 4160 Vac turbine generators and a distribution panel (including cables and connectors) per unit. The staff reviewed calculations 1-E-S-4KV-FLEX-001, "4.16 kV FLEX Event Phase 3 Cable Ampacity and Power Source Sizing," Rev. 0 and 2-E-S-4KV-FLEX-001, "4.16 kV FLEX Event Phase 3 Cable Ampacity and Power Source Sizing," Rev. 0. Based on its review, the NRC staff finds that the 4160 Vac equipment being supplied from the NSRCs will provide adequate power to enable CNP to maintain or restore core cooling, spent fuel pool cooling, and containment indefinitely following an ELAP.

3.2.4 Conclusions

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

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-2 and Appendix D summarize an acceptable approach consisting of three separate capabilities for the SFP cooling strategies. This approach uses a portable injection source to provide the capability for 1) makeup via hoses on the refueling floor capable of exceeding the boil-off rate for the design-basis heat load; 2) makeup via connection to 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 requires a vent pathway to vent steam from the SFP.

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

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

The ELAP causes a loss of cooling in the SFP. As a result, the pool water will heat up and eventually boil off. The licensee's response is to provide makeup water. The timing of operator actions and the required makeup rates depend on the decay heat level 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. The CNP has one SFP, shared by both units, located in the Auxiliary Building with the operating floor at the 650 ft. elevation.

3.3.1 Phase 1

Assuming that the initial SFP level is in accordance with that required by Technical Specifications (23 ft. over the top of irradiated fuel assemblies), the licensee determined that it would take approximately 49 hours to boil off the SFP water to a level requiring cooling or the addition of makeup to preclude fuel damage, conservatively assuming a dual unit, fresh core offload. Therefore, makeup to the SFP would not be required in Phase 1, although preparations would be made to provide makeup. The FIP states that moisture caused by evaporation or boiling will be removed from the Auxiliary Building by natural draft, which is established by operator actions to open the elevation 609 ft. Auxiliary Building crane bay roll up door and the SFP roof fire dampers in the Auxiliary Building roof (above elevation 650 ft.). The licensee also pre-stages hoses and a nozzle on the SFP operating floor before boiling begins, which is conservatively estimated to be in about 10 hours.

3.3.2 Phase 2

The licensee plans to initiate makeup to the SFP using the FLEX Lift pump, which also supports Phase 2 core cooling FLEX strategies by providing water to feed the SGs. After the SG makeup requirements are reduced or eliminated by switching core cooling to RHR cooling, which occurs prior to the 49 hour limit for SFP makeup, the FLEX lift pump would be used to draw water from Lake Michigan and deliver make-up water to the SFP using hoses to a pipe which will discharge into the SFP, or if necessary, to a fire protection monitor nozzle which would be mounted at elevation 650 ft. of the Auxiliary Building adjacent to the SFP. The FLEX Lift pump is capable of supplying the maximum boil-off rate of 115 gpm; if necessary the hose configuration allows for throttling of the supply flow as required.

3.3.3 Phase 3

The FIP states that the Phase 2 strategy would continue for an extended period and that no specific Phase 3 strategy is planned.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

The licensee' s Phase 1 SFP inventory control strategies rely on establishing a ventilation path to remove moisture caused by evaporation or boiling from the Auxiliary Building by natural draft and deploying hoses early in the ELAP event.

The CNP calculation PRA-SFP-HEAT-UP, "SFP Long Term Decay Heat Loads and SFP Heat Up Rates," demonstrates that the time required to reduce the SFP water to a level requiring cooling or the addition of makeup to preclude fuel damage is approximately 49 hours with a dual unit, fresh core offload. The staff noted that this scenario is unrealistic in that no actual simultaneous defueling of both units is planned; however, it provides a conservative estimate for the licensee to plan its response to an ELAP event.

During the audit, the staff noted that calculation PRA-STUDY-095, "Spent Fuel Pool (SFP) Heat Input and Removal Comparison," dated February 22, 2012, indicates that a loss of SFP cooling will result in a heat-up rate of about 0.875 °F/hr. The licensee indicated that the expected initial SFP temperature is approximately 90 °F. Thus, the licensee established a time constraint of 10 hours from the start of the ELAP event to establish a vent path and deploy hoses in order to avoid conflicts with other FLEX strategies and to ensure the SFP area remains habitable for personnel entry. During the audit, the staff noted that procedure, 12-OHP-4027-FSG-11, "Alternate SFP Makeup and Cooling," Rev. 1, provides guidance to align a vent pathway for the SFP by opening the Auxiliary Building crane bay roll-up door and roof fire dampers above the SFP operating floor. In addition, it was noted that a caution is provided to operators regarding habitability and radiation concerns from SFP boiling and reduced SFP level.

The licensee's Phase 2 and Phase 3 SFP inventory control strategies rely on the use of a FLEX Lift pump with suction from Lake Michigan which discharges to FLEX hoses that can either discharge into the pool or be connected to a fire protection monitor nozzle that delivers make-up water directly to the SFP, or to FLEX hoses that can be deployed to the hose connection at the 12-CS-290 valve. The staff noted that this valve is the fuel pool cooling and purification system valve located at elevation 617' CVCS Demineralizer Central Hallway (Auxiliary Building). The purification system piping will discharge the water into the SFP. The NRC staff's evaluation of the robustness and availability of FLEX connections points for the FLEX Lift pump is discussed in Section 3.7.3.1. The staff's evaluation of the robustness and availability of Lake Michigan 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. These instruments have initial local battery power with the capability to be powered from the 600 Vac FLEX DGs. The NRC staff's review of the SFP level instrumentation, including the primary and back-up channels, the display to monitor the SFP water level and environmental qualifications to operate reliably for an extended period are discussed in Section 4 of this safety evaluation.

3.3.4.2 Thermal-Hydraulic Analyses

The licensee's Overall Integrated Plan (OIP) [Reference 10] states that the bounding heat load for the SFP of 55.3 MBtu/hr is taken from Calculation/Report NSA-SFP-001, "Spent Fuel Pool (SFP) Cooling Analysis," Rev. 0. Based on this bounding heat load, the staff noted that the maximum boil off rate is equal to approximately 115 gpm as documented in EC-53212, Unit 1 FLEX Mitigation Strategies Overall EC, Attachment 25. The staff finds the licensee has considered the maximum design-basis SFP heat load for the site consistent with NEI 12-06, Section 3.2.1.6.

3.3.4.3 FLEX Pumps and Water Supplies

During the audit, the licensee provided calculation MD-12-FLEX-002-S, "DC Cook FLEX Core Cooling and SFP Makeup Hydraulic Analysis," Rev. 1, which determined the necessary pump performance criteria for the portable FLEX Lift Pump to support the licensee's SFP inventory control strategies. This calculation assessed numerous possible lineups based on such variables as suction sources, connection points and hose paths to determine adequate performance criteria for the FLEX Lift Pump providing makeup to the SFP via installed piping or hoses/monitor nozzle. The credited water source for Phase 2 and 3 during an ELAP event to support SFP inventory control is Lake Michigan.

During the audit, the NRC staff performed a walkdown of the licensee's SFP inventory control FLEX strategies and noted that the point of deployment for the portable FLEX pumps, hose routing and deployment connection points (primary and alternate) were consistent with the licensee's hydraulic analysis. The staff noted that the operating instructions for the FLEX Lift Pump are in procedure, 12-OHP-4027-FSG-311, "FLEX Lift Pump Operation," Rev. 0. One FLEX Lift Pump can support core cooling for both units or provide SFP makeup by lifting water from the forebay, which is connected to Lake Michigan by the North and South intake tunnels, and supplying a flow rate which exceeds the decay heat removal requirements. Operators will be present as necessary while the FLEX Lift Pump is in operation in order to control and maintain proper flow to support SFP inventory control based on information from the control room.

The NRC staff noted that the FLEX Lift Pump is credited to supply water for decay heat removal for the reactor core and for the SFP, but the licensee's hydraulic calculation, MD-12-FLEX-002-S, does not demonstrate the pump is capable of supporting both functions concurrently. The licensee's timeline shows core cooling being transitioned to RHR cooling before 48 hours after the event. The staff noted that calculation MD-12-FLEX-002-S demonstrates the ability to achieve at least 115 gpm for SFP makeup when the FLEX Lift pump only supports the SFP

FLEX strategy. The 115 gpm matches or exceeds the boil-off rate due to decay heat. The staff noted that calculation MD-12-FLEX-002-S did not demonstrate the ability to achieve 500 gpm to the SFP using spray nozzles (250 gpm per unit). The NRC staff requested that the licensee provide its basis for not providing the SFP spray flowrates recommended in NEI 12-06, Table D-3. In response, the licensee explained that although MD-12-FLEX-002-S did not directly demonstrate the SFP spray flowrates recommend by NEI 12-06, it is possible to use this calculation to show that the FLEX Lift Pump can provide the recommended spray flowrates when only supporting the SFP FLEX strategy. The licensee stated that in order to determine the spray flowrate capability that can be provided to the SFP by a FLEX Lift Pump, it is necessary to determine the pressure that would be applied to the inlet of the spray nozzle. Since the outlet pressure for the 5" hose from the FLEX Lift Pump to a 5" by 2.5" flow splitter (about 1800 ft. of hose) that would supply the SFP was not determined in the calculation, it was determined by using information from a similar 5" hose run from the FLEX Lift Pump to the FLEX Booster pump (about 2000 ft. of hose). The licensee indicated that the pressure provided by the FLEX Lift Pump at the suction of the FLEX Booster Pump calculated in MD-12-FLEX-002-S serves as an accurate predictor of the pressure delivered to the flow splitter located in the auxiliary building crane bay for water delivery to the SFP. The staff noted that this is reasonable because the deployed location of the FLEX Lift Pump is consistent in both scenarios and the deployed location of FLEX Booster Pump and flow splitter are at plant grade (609' elevation). In its assessment, the licensee considered the following: (1) adjustment for the pressure drop in the 2000 ft. run of 5" hose from 654 gpm (i.e., delivery to FLEX Booster Pump) to a value of 500 gpm (i.e., recommended SFP spray flowrate); (2) head loss in the hose run from the 609' elevation in the auxiliary building crane bay to the 650' elevation at the edge of the SFP; (3) head loss from the 150 ft. run of 2.5" hose used from the flow splitter to the monitor nozzle, noting that two hose runs and two monitor nozzles are used simultaneously; and (4) the pressure drop through the monitor nozzle. Based on the performance data for the hose monitors and the discharge pressure at the hose monitors, the staff finds it reasonable that the FLEX Lift Pump is capable of delivering at least 500 gpm of spray to the SFP.

Spray to the SFP is only needed if there is a leak in the SFP that lowers the water level below the level of the fuel assemblies. NEI 12-06, section 3.2.1.6, states that an initial SFP condition is that all boundaries of the SFP are intact; thus, the staff notes that the NEI 12-06 guidance to have spray available is a defense-in-depth measure, and the conditions that would require this capability (i.e., draining of the SFP and uncovering of the spent fuel) are extremely unlikely due to the robust construction of the SFP as a Seismic Category I structure. The NRC staff finds that the licensee has the capability to deliver 500 gpm of spray to the SFP. However, the staff finds that the licensee's capability does not fully meet the intent of NEI 12-06, as the capability is not independent of the need to provide makeup to the SGs. The 500 gpm spray flow cannot be achieved until after core cooling has been transitioned to RHR cooling. The licensee has another FLEX Lift Pump (the N+1 pump) in the FSB, but has not developed a strategy to use two Lift Pumps simultaneously. However, the staff finds that the licensee has a strategy to maintain or restore SFP cooling which will prevent damage to the fuel following a BDBEE, which meets the requirement of the EA-12-049 order. Therefore, the NRC staff finds that the dependence of the SFP spray flow rate on the makeup flow rate to the SGs is an acceptable alternative to NEI 12-06, as the licensee has demonstrated compliance with the order, and the staff concludes that the licensee could implement spray flow if necessary.

The staff noted that the specific procedures associated with the licensee's SFP inventory control strategies are contained in procedure 12-OHP-4027-FSG-11, "Alternate SFP Makeup and Cooling," Rev. 1, which provides guidance for restoring SFP level using an alternate source, and procedure 12-OHP-4027-FSG-1101, "Alternate SFP Makeup Equipment Deployment," Rev. 0, which provides guidance for deployment of Phase 2 hoses to supply makeup to the SFP via hose to a monitor nozzle or to the SFP demineralizer fill connection.

3.3.4.4 Electrical Analyses

The FLEX Lift pump used to supply makeup water to the SFP is diesel-driven. The equipment used to supply makeup water does not require electrical power.

3.3.5 Conclusions

The NRC staff concludes that the licensee has the three methods for SFP makeup stated in NEI 12-06, Table D-3, with the capability for a flow rate exceeding the boil-off rate based on a conservative plant-specific analysis of the fuel's decay heat and a capability to provide 500 gpm spray flow to the SFP. However, as discussed in section 3.3.4.3 above, the staff concludes that the licensee's capability does not fully meet the conditions of NEI 12-06, but does meet the requirements of the EA-12-049 order. The NRC staff finds that this is an acceptable alternative to NEI 12-06. 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, with an approved alternative, and adequately addresses 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 CNP units each have an ice condenser containment.

The licensee performed a containment evaluation, CN-SCC-13-004, "D.C. Cook ELAP Containment Environment Analysis", Rev. 0, which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of repowering a Containment Air Recirculation/Hydrogen Skimmer (CEQ) Fan 68 hours after an ELAP-inducing event. The calculation concludes that the containment parameters of pressure and temperature remain well below the respective UFSAR Section 5.2.2 design limits of 12 psig and 250 °F for more than 72 hours when this strategy is implemented.

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 CNP FIP states that the hydrogen igniters will be repowered within 12 hours following an ELAP.

3.4.1 Phase 1

The CNP containment analysis concludes that there are no Phase 1 actions required, as the containment pressure and temperature remain below their design limits.

3.4.2 Phase 2

The CNP FIP states that the upper and lower containment hydrogen igniter assemblies will be provided power through connections from either the 600 Vac, 500 kW FLEX DG, or the 480 Vac, 350 kW FLEX DG via the existing 480/600 Vac Outside Temporary Outage Power Transformer. The upper and lower containment hydrogen igniter assemblies are designed to maintain containment integrity by preventing hydrogen deflagration or detonation due to build-up of hydrogen gas in the event of core damage. 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. Action Item 8 of the FIP states that the hydrogen igniters will be repowered within 12 hours following an ELAP-inducing event.

3.4.3 Phase 3

The CNP FIP states that the NSRC-supplied 4160 Vac turbine generators would be available to provide power to the Train B CEQ Fan by the time active containment cooling is required. Initial containment cooling and depressurization would be accomplished by operating one CEQ Fan per unit and circulating the containment air volume through the ice condenser, cooling and depressurizing the containment. As stated above, this action would not be required for more than 72 hours; however, the licensee's calculation shows a more favorable containment response to the strategy of one CEQ Fan being repowered at 68 hours following an ELAP-inducing event.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

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

3.4.4.1.1 Plant SSCs

Sections 1.2.5 and 1.3.1 of the CNP UFSAR state that the ice condenser containment is a domed, steel-lined, reinforced concrete cylinder which is anchored to a reinforced concrete foundation slab. It is capable of withstanding a design pressure of 12 psig, and, as stated in Section 2.9.2, both the containment and the ice condenser are Seismic Class I structures. Section 2.9.5 further states that the containment structure has been designed for tornado loads. Finally, Table 5.3.2-1 shows that the total active volume of the containment is 1,179,636 cubic feet.

Section 5.5.3 of the UFSAR describes the Containment Air Recirculation/Hydrogen Skimmer System (CEQ System). The CEQ system is the only safety-related ventilation system in the containment. It consists of two redundant and independent systems located in the upper volume of containment whose function is to recirculate the containment atmosphere between the upper and lower compartments. The total system design air flow per train is 41,800 standard cubic feet per minute (scfm).

Based on these UFSAR qualifications, the ice condenser containment and the CEQ Fan credited in the strategy are robust, as defined by NEI 12-06, and would be available following an ELAP-inducing event.

3.4.4.1.2 Plant Instrumentation

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 12 hour 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 would be available. Included in this list of parameters are lower containment pressure readings for all four quadrants of the containment.

3.4.4.2 Thermal-Hydraulic Analyses

The licensee provided the staff with containment evaluation CN-SCC-13-004, "D.C. Cook ELAP Containment Environment Analysis", Rev. 0, 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 four cases of interest in evaluating the behavior of the containment. All four cases considered an additional 10 percent heat load above that which would be expected from the Reactor Coolant System to account for uncertainty.

Each of the four cases analyzed a 72-hour coping period. The results show that 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 CEQ fan is re-powered and provides the necessary differential pressure to open them.

Cases 1, 2, and 3 evaluated scenarios both with and without credit for the recently installed SHIELD® low leakage RCP seals. Additionally, there were combinations of various cooldown strategies (e.g. initiating RCS cooldown at 2 hours or 8 hours) coupled with and without the re-powering of the CEQ fans at 68 hours. The calculation concluded that each of these combinations resulted in containment pressure and temperature values being acceptable for at least 72 hours following an ELAP.

Case 4, however, was the model which specifically analyzed the licensee's credited strategy for core cooling (as described in Section 3.2.1.1.1) and re-powering one CEQ fan for containment cooling purposes. Specifically, this model incorporated credit for the recently installed SHIELD® low leakage RCP seals, initiation of RCS cooldown at 8 hours, and the re-powering of one of the

CEQ fans at 68 hours following an ELAP-initiating event. At hour 68, when the CEQ fan was turned on in the analytical model, the containment pressure was calculated to be approximately 5.8 psig and the temperature in the loop compartment was approximately 240 °F. After the CEQ fan was turned on, the doors to the ice condenser were opened and the containment pressure quickly returned to approximately atmospheric pressure and the temperature dropped to approximately 110 °F.

As stated in Section 3.2.1.2.2, the need may arise for the RCS to be vented to the containment atmosphere via the reactor vessel head vent system in order to facilitate injection of borated water using the FLEX boric acid pumps. The containment evaluation referred to above does not explicitly model the addition of the heat and mass associated with this venting operation; however, the leakage rates utilized in the non-SHIELD® RCP seal cases result in a containment response which ultimately bounds the response expected with SHIELD® RCP seals and the amount of reactor vessel head venting anticipated to be needed.

During an ELAP event, the containment heat up and pressurization is primarily driven by the leakage of the RCP Seals. In the design report, DAR-SCC-14-001, "ELAP Containment Environment GOTHIC Analysis Design Report for the D.C. Cook Unit 1 and Unit 2 Nuclear Plant", Rev. 0, it is shown that the expected leakage from SHIELD® low leakage RCP seals is considerably less than the non-SHIELD® RCP seals which, as demonstrated by the aforementioned analytical cases, ultimately showed acceptable results for at least the first 72 hours following an ELAP-initiating event with no other mitigating actions taken (e.g. starting a CEQ fan).

If the licensee implements their strategy appropriately and consistent with its FIP, the integrity of containment should be maintained.

3.4.4.3 FLEX Pumps and Water Supplies

For Phase 1 and Phase 2 with the unit operating within the boundary conditions of NEI 12-06, Section 2, the analysis demonstrates that there are no actions required to maintain pressure below the design limit of 12 psig for over 72 hours, which is adequate time for Phase 3 implementation.

During Phase 3, the NSRC-supplied 4160 Vac turbine generators would be available to provide power to one CEQ fan per unit. Initial containment cooling and depressurization would be accomplished by operating one CEQ fan per unit and circulating the containment air volume through the ice condenser, cooling and depressurizing the containment.

The staff noted that the licensee's containment integrity strategies do not rely on the use of FLEX pumps and associated water sources.

3.4.4.4 Electrical Analyses

The licensee has performed a containment analysis based on the boundary conditions described in Section 2 of NEI 12-06. Based on the results of this analysis, required actions to ensure maintenance of the containment integrity and required instrumentation function have been developed. However, there are no Phase 1 or Phase 2 actions that are required to maintain containment within its limits for over 72 hours. For Phase 2, the licensee will power the hydrogen igniter assemblies via the 600 Vac, 500 kW FLEX DG, or the 480 Vac, 350 kW FLEX DG via the existing 480/600 Vac Outside Temporary Outage Power Transformer. For Phase 3, containment cooling and depressurization is accomplished by operating one CEQ fan per unit and circulating air through the ice condenser. The CEQ fans will be powered by the 4160 Vac FLEX DG delivered from the NSRC. The licensee confirmed that the FLEX DGs have the necessary capacity to support the necessary equipment during Phases 2 and 3. The staff reviewed the analyses as discussed in Section 3.2.3.6 of this SE.

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 adequately addresses 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 BDBEE 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 site-specific external hazards leading to an ELAP and LUHS.

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

The licensee reviewed the plant site against 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 safety evaluation 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 20] (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies.

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," dated November 21, 2014). The Commission provided guidance in a Staff Requirements Memorandum (SRM) to COMSECY-14-0037 [Reference 21]. 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 45], 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 safety evaluations 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) dated March 6, 2015 [Reference 22], but the NRC staff has not completed a review of this report. The licensee developed its OIP for mitigation strategies in February 2013 [Reference 10] by considering the guidance in NEI 12-06 and its current design-basis hazards. Therefore, this safety evaluation 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 which 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 to reevaluate the seismic hazard at their site. The licensee submitted its seismic hazard and screening report (SHSR) dated March 27, 2014 [Reference 23], and the staff completed its review of the report, as documented by letter dated April 21, 2015 [Reference 24], and the results are discussed in Section 3.5.1 below. Therefore, this safety evaluation makes a determination based on the OIP and FIP, and notes the possibility of future actions by the licensee since the licensee's SHSR identifies a seismic hazard which exceeds the current design-basis seismic hazard.

The characterization of the specific 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 CNP site. In its SHSR, the licensee stated that per UFSAR Section 2.5.2, the design-basis earthquake (DBE) seismic criteria for CNP is two-tenths of the acceleration due to gravity (0.20g) peak horizontal ground acceleration and 0.133g peak ground acceleration acting vertically. It should be noted that the actual seismic hazard involves a spectral graph of the acceleration versus the frequency of the motion. Peak acceleration in the frequency range that affects structures, such as the numbers above, is often used as a shortened way to describe the hazard. The current NRC terminology for the DBE is the safe shutdown earthquake (SSE).

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

Based on the results of its SHSR, CNP screened-in for a risk evaluation, a high frequency evaluation, and a spent fuel pool evaluation. Electric Power Research Institute (EPRI) Report 3002000704 [Reference 26], 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 27]. 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 25]. This Augmented Approach ensures that installed plant equipment 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 Hz. This provides assurance of plant safety while the plant completes the seismic probabilistic risk assessment (SPRA).

The NRC staff completed its review of CNP's SHSR, as documented by letter dated April 21, 2015 [Reference 24]. 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 CNP 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 for CNP over the frequency range of 4 to 100 Hz.

By letter dated December 18, 2014, the licensee submitted its expedited seismic evaluation process (ESEP) report [Reference 31]. In the report, the licensee identified near-term modifications needed to the boric acid storage tanks' anchorage to raise the HCLPF above the RLGM. Further, more detailed risk evaluations are planned to be performed by the licensee. By letter dated August 25, 2015 (ADAMS Accession No. ML15232A411), 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 license'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 the FIP, the licensee stated that the design-basis flood results from a weather-driven seiche on Lake Michigan. The potential effect of such a seiche has been evaluated, as documented in CNP calculation MD-12-FLOOD-006-N, "Surge and Seiche, Cook Nuclear Plant Flood Hazard Re-evaluation." As documented in that calculation, combining a 10^{-6} exceedance peak base lake level of 582.3 feet with a 6.9 to 7.1 foot surge and seiche, and a 3.0 foot wave runup and setup, results in a peak Probable Maximum Surge and Seiche water surface elevation of 593.3 feet when converted to National Geodetic Vertical Datum of 1929 (NGVD29). The analysis determined that surge and seiche levels of Lake Michigan will remain below the current lakeside seawall level. Therefore, flooding of the plant site, which is at a general elevation of 609 feet, would not occur due to the reevaluated seiche. The plant configuration provides passive flood protection from the maximum seiche level and the portable FLEX equipment will be stored above the maximum seiche level. Given that a seiche is a relatively short duration event, the maximum seiche level is considered in the deployment of portable FLEX equipment.

The licensee submitted its FHRR, as noted in Section 3.5 above. The flood reevaluation considered the eight flood causing mechanisms and a combined effect flood required by the 50.54(f) letter. As discussed above, the reevaluation showed a flood level of 593.3 feet from the seiche and wave runup which is less than the current licensing basis (CLB) of 594.6 feet. The reevaluation of six other potentially flood causing mechanisms were deemed not relevant to CNP except for a flooding concern from local intense precipitation (LIP). The LIP event was not previously analyzed and thus is not considered in the CLB. Reevaluation of flooding resulting from LIP identified potential for water ingress into the TB and the AB. The licensee has committed to implementing interim measures to address the higher flooding levels relative to the current licensing basis. In addition, I&M is expected to complete a focused evaluation as described in Reference 45. The focused evaluation will be submitted as requested by the NRC. The NRC staff has not completed its review of CNP's FHRR.

During the audit process, the licensee addressed the potential impact of ground water in-leakage and any potential impacts from failure of large internal flooding sources. The licensee stated that the maximum lake levels are below the elevation of equipment expected to be utilized in the FLEX strategies and there are no other cooling basins for non-safety related cooling systems on site. The licensee further stated that there is no equipment utilized in the FLEX strategies that relies on ac power to mitigate ground water.

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 first part of the evaluation of high wind challenges is determining whether the site is potentially susceptible to different high wind conditions to allow characterization of the applicable high wind hazard. The second part is the characterization of the applicable high wind threat.

The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06, Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical

Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009); if the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 miles per hour (mph) exceeds 10^{-6} per year probability, the site should address hazards due to extreme high winds associated with hurricanes.

The screening for high wind hazards associated with tornadoes should be accomplished by comparing the site location to NEI 12-06, Figure 7-2, from U.S. NRC, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461, Rev. 2, February 2007; if the recommended tornado design wind speed for a 10^{-6} per year probability exceeds 130 mph, the site should address hazards due to extreme high winds associated with tornadoes.

In its FIP, the licensee stated that Figure 7-2 from NEI 12-06 was used for assessment of the high wind hazard. It was stated that the CNP site is in Region 1 of this figure resulting in a FLEX design wind speed of 200 mph. The licensee also stated that the FLEX storage building was designed for protection against the tornado-generated missiles listed in UFSAR Table 5.1-1; a 4000 pound passenger car moving along the ground at 50 mph, a piece of wood decking (12 feet by 12 feet by 4 inches, weighing 450 pounds) traveling at 200 mph, and a piece of corrugated sheet siding (4 feet by 4 feet weighing 100 pounds) traveling at 225 mph.

The NRC staff compared the documented location for CNP with NEI 12-06, Figure 7-1 and verified that the site is in an area that has a frequency of recurrence of hurricanes with wind speeds in excess of 130 mph with less than 10^{-6} per year probability, which would screen out the high wind hazard due to hurricanes, leaving only the high wind hazard due to tornadoes, which was considered by the licensee in developing the mitigation strategies.

The licensee has appropriately screened in the high wind hazard and characterized the hazard in terms of wind velocities and wind-borne missiles.

3.5.4 Snow, Ice, and Extreme Cold

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

In its FIP, the licensee further described that Figure 8-2 in NEI 12-06 is a Maximum Ice Storm Severity Map based on a database developed by EPRI which summarized ice storms that occurred in the United States from 1959 to April 1995. Using Figure 8-2, the licensee determined that the CNP site is located in an ice severity level 5 region, "Catastrophic destruction to power lines and/or existence of extreme amount of ice".

In its updated FIP [Reference 44], the licensee stated it had evaluated the storage and functionality of the FLEX equipment to outdoor temperatures of -20 °F, which is appropriate for the plant's location.

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 NEI 12-06, Section 9 states that all sites will address high temperatures. Virtually every state in the lower 48 contiguous United States has experienced temperatures in excess of 110 °F. Many states have experienced temperatures in excess of 120 °F. In this case, sites should consider the impacts of these conditions on deployment of the FLEX equipment.

In its FIP, the licensee stated that records indicate that the highest temperature recorded for the nearest municipality, Bridgman, Michigan, was at 103 °F in July 1999.

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 adequately addresses 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

Most of the FLEX equipment will be stored in the newly constructed FSB, which is located within the Owner Controlled Area but outside the Protected Area (PA). The building is a stand-alone, reinforced concrete structure, consisting of a reinforced concrete slab-on-ground foundation and reinforced pre-stressed pre-cast concrete walls and roof members. The building has the following features:

- Two steel personnel entry doors, one each on the north and south wall.
- One large motor operated horizontal steel rolling door for equipment entry and exit. The door can also be opened via hand crank, or by use of the installed motor powered by a portable generator, or by manually applying horizontal force to the door.

- One knock-out opening in the concrete wall which can be used in an emergency for equipment entry and exit in case the equipment door is rendered non-functional.
- Ramps for the vehicles to enter and exit through the rolling door or the knock-out panel openings.
- A manually operated vehicle barrier in front of the knock-out panel to protect it from rolling vehicles.
- Concrete blocks around the building to protect the building against accidental rolling vehicles.

The FSB is powered from an existing 12 kV line via a dedicated transformer bank. The power feed is via overhead cabling. A 100 amp generator receptacle is installed inside the building for connection of a backup source of power in the event of loss of external power.

Portable equipment stored in the FSB includes: two diesel-driven FLEX Lift pumps, two diesel-driven FLEX Booster pumps, two FLEX 250 kW DGs to power the boric acid FLEX pumps, two 500 kW 600 Vac DGs, one 350 kW 480 Vac DG with a portable 480/600 Vac step up transformer, two diesel-driven blended RCS makeup pumps, one diesel fuel transport trailer, two pickup trucks with snow plows used as tow vehicles and for debris removal, and other miscellaneous portable debris removal equipment. The 350 kW DG provides the N+1 function for the two 500 kW DGs. The hoses and cabling needed to connect the FLEX equipment to the plant tie-in points are also stored in the FSB.

FLEX equipment is also stored outdoors outside the protected area. The FLEX debris removal equipment includes two large front-end loaders which are stored in two separate outside locations. One front-end loader is stored near the FSB, and the other is stored near the independent spent fuel storage installation (ISFSI) area.

FLEX equipment is also stored inside the auxiliary building and the turbine building. Portable equipment stored in the auxiliary building includes: three portable electric-powered FLEX boric acid pumps mounted on mobile carts, an "E-Cart" containing a 480/120 Vac transformer, mixing manifolds and other miscellaneous hoses, cables and tools. Two 26 kW DGs with a battery powered equipment mover are stored under the main generator in the turbine building.

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

3.6.1.1 Seismic

The licensee stated in its FIP that the portable FLEX equipment stored in the FSB is protected against the hazard of an SSE. Because it is outside the PA and does not affect the safety of the plant, the FSB was designated as a non-safety-related building. However, special requirements were applied. The building was designed to meet Seismic Class I design requirements, which meets NEI 12-06 guidance. The building was designed to meet CNP site-specific seismic spectra corresponding to the DBE.

The licensee stated in its FIP that an evaluation determined that tie-downs for securing major equipment within the FSB are not required. This conclusion was reached by evaluating the most limiting component, the 500 kW generators, for sliding and overturning. For storage of FLEX equipment in areas other than the FSB, walkdowns were performed to ensure there would be no adverse impacts to surrounding safety-related equipment, and to ensure that existing plant equipment would not damage the staged FLEX equipment during a seismic event.

The auxiliary building and the portion of the turbine building supporting the mitigation strategies are Seismic Class 1 structures, which are designed to withstand the DBE.

3.6.1.2 Flooding

The licensee stated in its FIP that the FSB has a building floor elevation of 625 feet 6 inches, which is well above the CLB flood elevation level of 594.6 feet and the general site elevation of 609 feet. The turbine building and auxiliary buildings are also above the CLB flood elevations. However, as noted in section 3.5.2 above, the flood reevaluation identified potential for water ingress into the turbine building and auxiliary building resulting from local intense precipitation. The licensee has committed to implementing interim measures to address these findings. These measures include blocking various floor drains and sealing affected gaps and penetrations. In addition the licensee will evaluate the need for additional administrative measures to preclude or minimize water ingress.

3.6.1.3 High Winds

The licensee stated in its FIP that the FSB was designed for tornado wind loads resulting from a maximum tornado wind velocity of 360 mph (a tornado with a forward progression of 60 mph with rotational wind speed of 300 mph) and a coincidental pressure drop of 3 psi applied within three seconds, which is consistent with the CNP UFSAR. The building was designed for protection against the following tornado-generated missiles per UFSAR Table 5.1-1:

- Bolted wood decking- 12 ft. x 12 ft. x 4 in., 450 lbs. traveling at 200 mph.
- Corrugated sheet siding- 4 ft. x 4 ft. 100 lbs. traveling at 225 mph.
- Passenger car- 4000 lbs. traveling along the ground at 50 mph.

The licensee stated that the two front-end loaders stored outdoors are sufficiently separated such that there is assurance that at least one of the front-end loaders would survive the applicable site hazards, such as a tornado. During the audit, the licensee stated that the front-end loaders are stored approximately 1500 feet apart and roughly perpendicular to the predominant tornado path. In addition, one diesel fuel transport trailer is stored near the ISFSI area, another one is stored near the switchyards, and the third is stored inside the FSB.

The auxiliary building and the portion of the turbine building supporting the mitigation strategies are designed to withstand high winds and tornado borne missiles.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

The licensee stated in its FIP that the FSB was designed for snow load in accordance with the Michigan Building Code. All other design loads, such as the dead load, live load and load combinations were in accordance with ASCE 7-05, "Minimum Design Loads for Buildings and Other Structures" or ACI 318-63, "Building Code Requirements for Reinforced Concrete."

The licensee stated that the FSB ventilation system is designed to limit the minimum internal temperature to 50 °F based on a 0 °F outdoor air temperature. The system consists of a single exhaust fan, fixed and manually operated louvers, and two 15 kW electric heaters. In its updated FIP, the licensee further stated that the additional heat source inside the building of 17.1 kW in the engine block heaters was conservatively excluded. The licensee concluded that this additional heat would be expected to increase the internal temperature of the FSB to over 55 °F with an external temperature of -20 °F.

In the updated FIP the licensee also stated that the FLEX diesel generators are capable of operating in the extreme low temperature of -20 °F.

In its OIP, the licensee stated that storage/protection of equipment from high temperature hazard would be provided in storage structures that will be ventilated to allow equipment to function. Active cooling systems are not required as normal room ventilation will be utilized. The FLEX equipment was purchased with the capability of operating in the high temperatures described in Section 3.5.5.

3.6.2 Reliability of FLEX Equipment

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

Based on the number of portable FLEX pumps, 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.

For core cooling in Phase 2, the licensee has developed multiple strategies for adding water to the SGs. One strategy for adding water to the SGs is by using the FLEX lift pump drawing water from Lake Michigan and discharging to the suction of the TDAFW pump in each unit. The TDAFW pumps deliver the water to the four SGs in each unit using the existing TDAFW pump discharge header piping. Another strategy uses the FLEX lift pump discharging to the suction of two FLEX booster pumps, each aligned to deliver water to the SGs in one unit. In its updated FIP, the licensee described another strategy using one lift pump in combination with one FLEX booster pump which feeds two SGs in Unit 1 and two SGs in Unit 2 concurrently. The licensee stated that a single booster pump has the capacity to support decay heat removal in both units thus allowing the two FLEX booster pumps to meet the recommendation for having N+1 equipment in accordance with the NEI 12-06 guidelines.

3.6.3 Conclusions

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

3.7 Planned Deployment of FLEX Equipment

3.7.1 Means of Deployment

In its FIP, the licensee indicated that the debris removal equipment includes two pickup trucks equipped with snow plows, stored in the FSB, and two large Caterpillar 930H front-end loaders (or similar machinery) normally stored outdoors near the FSB and ISFSI areas respectively. The front-end loaders are available to deal with more significant debris conditions. Other miscellaneous debris removal equipment such as chain saws, a plasma cutter, power saws, a hydraulic spreader and cutter, bolt cutters and other miscellaneous tools are stored inside the FSB.

The larger FLEX portable equipment, such as pumps and generators, are trailer mounted and would be deployed by the two pickup trucks after debris removal was accomplished. The most limiting component weight of 19,485 lbs. (the 500 kW diesel-driven generators) was considered when specifying the towing capability of the pickup trucks.

Additionally, a battery powered equipment mover is stored in the turbine building for movement of equipment, such as the 26 kW DGs stored there.

The licensee stated that the deployment of the debris removal equipment and the FLEX equipment from the FSB is not dependent on electric power. The building horizontal steel rolling door can be manually operated via a hand crank.

3.7.2 Deployment Strategies

In its FIP, the licensee indicated that pre-determined, preferred haul paths have been identified and have been reviewed for potential soil liquefaction. The haul paths evaluated were from the FSB to the point of deployment within the PA and from equipment staging area "B", where the NSRC equipment will be delivered, to the point of deployment in the PA. The soil liquefaction evaluation determined estimated haul path settlements of up to 3 inches, which may slow traffic, but should not impair transport vehicles from proceeding to the power block area.

For the core cooling strategy (which requires makeup water to the SGs), a lift pump which is stored in the FSB would be deployed to the circulating water intake structure (forebay) where a suction hose is lowered through removable cover plates and manholes to draw water from Lake Michigan. A booster pump, which is also stored in the FSB, would be deployed near a Unit 1 or Unit 2 auxiliary building access port in close proximity to an MDAFW pump discharge header tie in location for the selected unit. The access port is selected based on availability following the event. The lift pump could be aligned to feed both units TDAFW pumps using hoses to the ESW supply pipe or feed the booster pump using hoses.

For RCS inventory and reactivity control, two electrically-driven boric acid pumps, stored in the auxiliary building on carts, are moved into position to connect into the boric acid transfer pump suction header and into the primary or alternate connection points for injecting into the RCS. One 250 kW FLEX DG to power these pumps will be deployed from the FSB and staged outside the auxiliary building crane bay rollup door.

For SFP cooling, a hose is run from the lift pump staged at the circulating water intake structure through the auxiliary building roll up door and up to the spent fuel pool or to the spent fuel pool demineralizer resin fill line connection.

The South Bend International Airport would be the single offsite staging area "C" for equipment from either the Memphis or Phoenix NSRC facility. The South Bend International Airport is approximately 20 air miles from CNP. Primary and secondary land routes from staging area "C" to CNP have been identified. Helicopter delivery to the site from staging area "C" would be used if all land routes were impassable.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

As described in its FIP, one of the licensee's Phase 2 core cooling strategies relies on a FLEX Lift pump taking suction from Lake Michigan and discharging into ESW supply piping aligned to the suction of the TDAFW pumps via a hose connection installed on the ESW supply piping in both units by a plant modification. These new connections are located in the portion of the turbine building (which is shared by both units) that is seismically robust, protected from tornadoes, and not susceptible to flooding. The licensee stated that the Engineering Changes replaced a blind flange on the existing 6 in. ESW piping with a 6 in. x 4 in. reducing flange, elbow, and piping on both units to allow connection of hoses from the FLEX Lift pump.

As described in the FIP, another of the licensee's Phase 2 core cooling strategies relies on a FLEX Lift pump taking suction from Lake Michigan and discharging to a FLEX Booster pump. The discharge of the FLEX Booster pump can be routed to the following points:

- To a newly installed connection point for flow to SG1 and SG4, which involves a 4 in. diameter pipe segment, isolation valve 1/2-FW-214, and pipe cap on the AFW System discharge piping from the West MDAFW pump in the Auxiliary Building. This connection point can also be used to provide flow to SG2 and SG3 in the opposite unit using existing installed cross-connect piping.

- To existing 1" SG drain connections on the main feedwater header to SG1 through SG 4. The staff noted that the details of these connections to the 1" drain connections are documented in EC-0000053212 and EC-0000053213.

The NRC staff noted that these connections for the Phase 2 core cooling strategies are located in the Auxiliary Building, which is safety related and provides protection of the connection points against all applicable hazards.

For Phase 3 core cooling strategies the licensee will place the RCS on RHR system cooling. The ESW flow to cool the CCW heat exchangers is achieved by removing the cover of one ESW pump discharge duplex strainer on both units and replacing it with a FLEX strainer lid adapter equipped with hose connections to accept discharge from the NSRC raw water pump. The FLEX strainer lid adapters and hose manifolds were fabricated for both units and are stored in the FSB.

For Phase 2 RCS inventory control and boration strategies, the licensee stated in its FIP that the FLEX boric acid pumps take suction from the BASTs via a new connection in the common suction header of the existing BAST pumps. The primary injection point to the RCS is a new connection at the existing reciprocating pump discharge header and the alternate connection points are through existing SI pump discharge piping vent and drain connections using a portable SIS manifold. The staff noted that the primary and alternate connection points that support Phase 2 RCS inventory control and boration strategies are located in the auxiliary building, which is safety related and provides protection of the connection points against all applicable hazards.

For Phase 2 SFP inventory control strategies, makeup water would be provided by the FLEX Lift pump using hoses and a fire protection monitor nozzle mounted adjacent to the spent fuel pool at the 650 foot elevation of the AB. The NRC staff notes that procedure 12-OHP-4027-FSG-1101 provides for an alternate connection point for SFP make-up through the spent fuel pool demineralizer resin fill line. Furthermore, the staff noted in this procedure that the discharge hose from the FLEX Lift pump would be routed through the AB roll up door either to the refueling deck or to the demineralizer resin fill line. The staff noted that both connection points are located in the AB, which is safety related and provides protection of the connection points against all hazards.

3.7.3.2 Electrical Connection Points

3.7.3.2.1 600 Vac FLEX DG (500 kW) (Primary Strategy or N Strategy)

In its FIP, the licensee stated that its primary strategy uses two 600 Vac FLEX DGs (500 kW), one of which will be deployed near the main transformer for each unit. FLEX cables will be routed through wall penetrations in the auxiliary building and connected to the Unit 1, 600 V switchgear bus 1-11D, breaker compartment 1-11D2 and Unit 2, 600 V switchgear bus 2-21D, breaker compartment 2-21D2. The auxiliary building is safety related and provides protection of the connection points against all applicable hazards.

3.7.3.2.2 480 Vac FLEX DG (350 kW) (Alternate Strategy or N+1 Strategy)

The alternate strategy is used if one of the 600 Vac DGs or its connection point fails or is unavailable. In its FIP, the licensee stated that its alternate strategy uses a 480 Vac FLEX DG (350 kW) which will be deployed near the RWST and Reserve Feed Transformers for Unit 1 and near the main transformer for Unit 2; the FLEX 480/600 V transformer will be deployed near the RWST and Reserve Feed Transformers for Unit 1 and near the RWST and Containment for Unit 2; and the E-cart containing the 480/120 V transformer will be deployed near the associated unit's isolimiter transformers in the auxiliary building. For powering the safety injection accumulator valves, FLEX cables will be routed from the FLEX DG to a trailer-mounted

480/600V transformer to Unit 1 MCCs 1-EZC-A/B/C/D and Unit 2 MCCs 2-EZC-A/B/C/D. For powering the control room instrument distribution panels (CRIDs), FLEX cables will be routed from the FLEX DG to an E-cart containing a 480/120 V transformer and from there to the isolimiter transformer outlet panel. The auxiliary building is safety related and provides protection of the connection points against all applicable hazards.

3.7.3.2.3 NSRC 4160 Vac FLEX DG (1 MW)

In its FIP, the licensee stated that in Phase 3 two 1 MW turbine NSRC FLEX generators would be connected in parallel for each unit and deployed near the north wall (for Unit 1) and east wall (for Unit 2) of the auxiliary building. To restore power to the Unit 1 Train B engineered safety systems, cables between the NSRC-provided paralleling switchgear and the Unit 1 4160 Vac RCP bus would be routed through penetrations on the north wall of the Auxiliary Building and terminated in cubicle 1-1A2. To restore power to the Unit 2 Train B engineered safety system, cables between the NSRC-provided paralleling switchgear and the Unit 2 4160 Vac RCP bus 2A would be routed through six new penetrations on the east wall of the Auxiliary Building and terminated in cubicle 2-2A2. The NSRC generators would repower buses and associated MCCs which supply CCW, RHR, ESW, and Control Room Cooling equipment and auxiliaries.

3.7.4 Accessibility and Lighting

In its FIP, the licensee stated that it has conducted reviews which documented consistency of the CNP FLEX validation actions with those prescribed in the NEI document titled "FLEX Validation Process." The reviews were conducted to determine if there would be adequate resources for simultaneous implementation of FLEX strategies at both units within the required constraints identified for Phases 1 and 2, and included consideration of lighting for personnel to perform the required actions. The reviews identified actions for which headlamps, flashlights, and portable lighting may be needed. The availability of headlamps, flashlights, and portable lighting is identified in plant procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," Rev. 1. This procedure states that some portable lighting is stored in the new FSB.

In its OIP, the licensee stated that lighting is required for initial operator access in the plant to implement actions associated with plant procedures. The designed emergency lighting will not be available due to being stripped from the batteries in order to extend battery capability. Available lighting will be the Appendix R light units (with built-in 8 hour batteries) and the portable lighting that personnel can carry, such as headlamps and flashlights. During Phase 3, portable generators will be utilized to provide power to available installed emergency ac lighting. Portable lighting units will be deployed externally as needed.

3.7.5 Access to Protected and Vital Areas

In its OIP, the licensee stated that contingencies are in place to provide access to areas required for the ELAP response if the security system is without power.

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee stated that the general coping strategy for refueling the diesel-powered FLEX equipment, i.e., pumps and generators, is to draw fuel oil from the two emergency diesel generator fuel oil storage tanks (FOSTs) located at the site, both of which are below-ground tanks. Technical Specification (TS) Surveillance Requirement 3.8.3.1 requires that at least 46,000 gallons of fuel oil be maintained in an FOST when either of the emergency diesel generators (EDGs) associated with an FOST is required to be operable. The TS requires diesel fuel oil sampling and testing in accordance with applicable ASTM Standards. Thus, based on its review, the NRC staff finds that the licensee has addressed management of fuel oil quality in its two FOSTs to ensure that the diesel FLEX equipment can be expected to be supplied with quality fuel oil. The FLEX equipment will be stored with fuel in their individual tanks. The staff's review of the licensee's maintenance and testing of FLEX equipment is documented in SE Section 3.13.

In its FIP, the licensee stated that the projected fuel oil usage for FLEX equipment is not expected to exceed 232 gallons per hour (gph), therefore, one FOST contains sufficient fuel oil for greater than eight days of continuous use. The licensee stated in its compliance letter dated June 16, 2015, that a fuel consumption study was conducted which estimated the total run time with available on-site fuel (both FOSTs) to be approximately 344 hours or approximately 14 days. The NRC staff noted that the licensee assumed that all major on-site diesel powered FLEX equipment (i.e., N and N+1) was running continuously at full load, which is conservative because the spare set of FLEX equipment is not expected to operating at the same time as the N set, the use of all diesel-powered FLEX equipment at the same time is not expected and FLEX equipment is not expected to be run at full load for the duration of the event. The staff noted in procedure 1-OHP-4027-FSG-5, "Initial Assessment and FLEX Equipment Staging," that specific direction is provided to consult with the Emergency Director to obtain diesel fuel from offsite sources before onsite supplies are used up. Although the diesel-powered equipment from the NSRC requires a larger fuel supply than the onsite FLEX equipment, there is enough fuel oil at the site that there is not an immediate need for resupply. Thus, the staff finds it reasonable that the licensee has sufficient time to obtain fuel oil from off-site to support Phase 2 and Phase 3 diesel-powered FLEX equipment.

Since the licensee's FLEX strategies are expected to transition from Phase 2 to Phase 3 no later than 72 hours after the event, one FOST would provide sufficient fuel capacity. For Phase 3, the SAFER response team would provide for alternate means (i.e. fuel transfer equipment and air-lift fuel containers) of delivery of fuel to the site until normal site access is available. Once normal site access has been restored, the licensee would provide for bulk delivery of fuel. If normal access was not restored within 24 hours, the licensee would provide for delivery of bulk fuel to Staging Area "C". From there, fuel containers could be air-lifted to the site.

The FLEX equipment includes three 500 gallon diesel fuel transport trailers and two diesel-powered fuel transfer pumps with associated hoses and fittings. The diesel fuel transport trailers are stored in three diverse locations (one inside the FSB, one near the ISFSI area, and one at a site equipment storage area near the switchyards).

Both FOSTs are buried below ground and the connection point for accessing the fuel is 18 inches above the general grade of 609 feet. The FOSTs and the fill connection are not susceptible to flooding. The NRC staff finds the licensee has a protected source of fuel oil consistent with NEI 12-06, Section 3.2.1.3, and has the ability to access the fuel oil during an ELAP event.

Procedure 12-0HP-4027-FSG-511, "FLEX Equipment Refueling Operation," provides direction to 1) move diesel fuel from the underground emergency diesel FOSTs, using the fuel transfer pumps, to the mobile fuel transport trailers, and 2) from the mobile fuel transport trailers to various FLEX equipment diesel engine fuel tanks.

The NRC staff noted that in the licensee's procedure, 12-OHP-4027-FSG-511, "FLEX Equipment Refueling Operation," it specifically identifies Phase 2 portable diesel powered FLEX equipment, the quantity available for each piece of equipment, the fuel consumption rate, fuel tank size and expected run time per tank of fuel in a tabular format. The staff noted that this provides a quick reference to operators as to the amount of time a piece of FLEX equipment can be expected to operate per tank of fuel. In addition, a scheduling tool is provided to operators to mark whether a piece of FLEX equipment is deployed, the deployed location, time of last refueling and next refueling check, which provides the licensee the ability to track refueling operations and refueling needs of its diesel powered FLEX equipment to ensure equipment does not run out of fuel oil. The staff noted that in licensee procedures 12-OHP-4027-FSG-311, "FLEX Lift pump Operation," 12-OHP-4027-FSG-312, "FLEX Booster pump Operation," 12-OHP-4027-FSG-411, "FLEX 500 KW DG Operation," 12-OHP-4027-FSG-412, "FLEX 250/350 KW DG Operation," and 12-OHP-4027-FSG-413, "FLEX 26 KW DG Operation, operators are directed to check fuel levels and request refueling at ½ tank to ensure uninterrupted operation of diesel powered FLEX equipment. Based on the guidance provided in these procedures, the staff finds it is reasonable that diesel-powered FLEX equipment will be refueled to ensure uninterrupted operation to support the licensee's FLEX strategies.

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 adequately addresses the requirements of the order.

3.8 Considerations in Using Offsite Resources

3.8.1 D.C. Cook SAFER Plan

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

By letter dated September 26, 2014 [Reference 28], 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.

3.8.2 Staging Areas

The licensee stated in its FIP that in the event of a BDB external event and subsequent ELAP/LUHS condition, equipment will be moved from an NSRC to a local assembly area designated as Staging Area "C". For CNP, Staging Area "C" is the South Bend International Airport. From there, equipment would be taken by ground transportation to the CNP site and staged at the large parking lot east of the CNP training building designated as Staging Area "B". Equipment can be delivered from Staging Area "C" by helicopter if ground transportation is unavailable.

Communications would be established between CNP personnel and the SAFER team via satellite phones, and required equipment would be moved to its final location inside the protected area of the plant, designated as Staging Area "A", as needed. The first arriving equipment would be delivered to the site within 24 hours from the initial request. The order in which equipment is delivered is identified in the "SAFER Response Plan for Donald C. Cook Nuclear Plant".

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 adequately addresses the requirements of the order.

3.9 Habitability and Operations

3.9.1 Equipment Operating Conditions

3.9.1.1 Loss of Ventilation and Cooling

The licensee's Phase 1 core cooling FLEX strategy relies on the TDAFW pump as the motive force for providing water to the SGs. The FIP states that procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," directs personnel to open the doors to the affected TDAFW Pump Room, which increases the volume of air available to dissipate heat. In addition, the procedure directs personnel to install a temporary portable ventilation fan with an exhaust duct routed out of the room and outside the hallway to the TB to provide supplemental cooling and a flow path to exhaust heated air out of the TDAFW pump room. During the audit, the NRC staff noted in DB-12-AFWS, "Design Basis Document for the Auxiliary Feedwater System," Rev. 5, and MP-4030-001-001, "Impact of Safety Related Ventilation on the Operability of Technical Specification Equipment – Turbine Driven Auxiliary Feedpump Rooms Ventilation," Rev. 15, that the maximum temperature for TDAFW pump survivability is 133.6 °F and that the steady-state temperature in the TDAFW pump room with loss of ventilation and with doors closed is 131 °F for the 4-hour station blackout (SBO) coping, respectively. In addition, during the audit, the staff noted that DB-12-AFWS indicates that the discharge from the TDAFW pump provides cooling water to the turbine driver's governor oil and lube oil coolers. Based on these licensee documents, and procedures for opening up TDAFW pump room doors and setting up portable exhaust fans, the staff finds it reasonable that the TDAFW pump will remain available during an ELAP event with loss of normal ventilation.

In the FIP, the licensee stated that a GOTHIC model was used to show the maximum temperature in the main control room during an ELAP is 117 °F with temporary fans installed. Procedure 1/2-OHP-4023-ECA-0.0, "Loss of All AC Power," directs the licensee to open main control room cabinet doors within 30 minutes and procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," Rev. 1, directs installation of temporary fans in response to a loss of control room ventilation and provides a sketch of the approximate fan locations. Based on opening cabinet doors and installing temporary fans, the staff finds it reasonable that the equipment in the main control room will not be impacted by the loss of ventilation.

The NRC staff also walked down the vital battery rooms at CNP to confirm the adequacy of the battery room ventilation. In the FIP, the licensee stated that the expected maximum temperature in the vital battery rooms during Phase 1 with loss of ventilation is 110 °F. The licensee plans to restore normal battery room ventilation when the batteries are charging during Phase 2 and 3, except for the Unit 2 N Train battery room fan during Phase 2 (refer to section 3.2.3.6 above). The fans would draw air as designed through the battery rooms and the room temperatures would trend toward the ambient air temperature of the auxiliary building interior. In procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," Rev. 1, the licensee also plans to open battery room doors and set up portable exhaust fans if required. As a result of its review, the NRC staff did not identify any issues with ventilation of the battery rooms.

During the audit, the staff requested that the licensee discuss the design and procurement of FLEX equipment regarding the ability to operate in a high temperature environment. The licensee stated in its response that FLEX equipment will be procured and designed for the expected environmental conditions and/or ventilation/cooling will be implemented. The staff noted that the Unit 1 and Unit 2 Engineering Changes EC-0000053212, "Unit 1 FLEX Mitigation Strategies Overall EC," and EC-0000053213, "Unit 2 FLEX Mitigation Strategies Overall EC," provides the supporting documents from the respective commercial vendors related to the FLEX equipment specifications.

There are three FLEX Boric Acid Pumps stored in the auxiliary building at elevation 587 ft. The pumps are mounted on mobile carts and can be moved to the vicinity of the CVCS Reciprocating Charging Pump or SIS rooms at that same building elevation. The FLEX Boric Acid Pumps are powered by a FLEX 250 kW DG which has a deployed location outside the auxiliary building crane bay rollup door. During the audit, the licensee identified the deployed location of the FLEX Boric Acid Pumps and the staff noted it is not in a confined area of the auxiliary building, and that the auxiliary building is temperature controlled prior to the ELAP event. Based on the intermittent use of the FLEX Boric Acid Pumps and their deployed location within the Auxiliary Building, the staff finds it reasonable that the FLEX Boric Acid Pumps will not be impacted by the loss of ventilation.

3.9.1.2 Loss of Heating

In NEI 12-06, Section 3.2.2, guideline (12) states that heat tracing is used at some plants to ensure cold weather conditions do not result in freezing important piping and instrumentation systems with small diameter piping. The staff noted that the water sources associated with reactor core cooling and RCS boration and inventory control FLEX strategies may be impacted by extreme cold weather.

The licensee's Phase 1 and Phase 2 core cooling FLEX strategies rely on the CST and Lake Michigan as the credited water sources to supply the SGs, respectively. The licensee stated that the CSTs and interconnecting piping outside the auxiliary building are insulated and adequately protected from extremes of hot and cold weather. The volume and initial temperature of the CST contents and associated piping would preclude the significant loss of heat required for freezing upon loss of all ac power within the Phase 1 timeframes. Furthermore, Lake Michigan is not susceptible to large scale freezing and the CW intake tunnels maintain communication with Lake Michigan below the anticipated level of winter ice cover. The piping which would be used is not small diameter and the required flow rates from the FLEX Lift pump would be expected to preclude freezing of its discharge path. During the audit, the NRC staff noted that the licensee determined in LTR-FSE-13-69, Rev. 0, that the CST volume and initial temperature prior to an ELAP would preclude freezing during the timeframe that the CST is going to be used as a suction source for the TDAFW pump. Based on the initial temperature and volume of the CSTs, insulation of the piping located outside, and the timeframe for using the CSTs in the licensee's FLEX strategies, the NRC staff finds it reasonable that the CSTs will remain available during an ELAP event and are not susceptible to freezing concerns.

The licensee's RCS inventory control FLEX strategies (Modes 1-4) rely on the BASTs as the credited water sources. The FIP states that the BASTs, which are located in the temperature-controlled auxiliary building, are maintained at 105 °F minimum by automatically controlled immersion heaters and the boric acid concentration solubility limit is 53 °F at the specified weight percent (6550 ppm). Based on the initial temperature and volume of the BASTs, the location of the tanks and piping within the auxiliary building, and the timeframe for using the BASTs in the licensee's FLEX strategies, the staff finds it reasonable that the BASTs remain available during an ELAP event and are not susceptible to freezing concerns.

There are three FLEX Boric Acid Pumps stored in the auxiliary building at elevation 587 ft. The pumps are mounted on mobile carts and can be moved to the vicinity of the CVCS Reciprocating Charging Pump or SIS rooms at that same building elevation. The FLEX Boric Acid Pumps are powered by a FLEX 250 kW DG with a deployed location outside the auxiliary building crane bay rollup door. During the audit, the licensee identified the deployed location of the FLEX Boric Acid Pumps and the staff noted it is not in a confined area of the auxiliary building, which is temperature controlled prior to the ELAP event. Based on the intermittent use of the FLEX Boric Acid Pumps and its deployed location within the auxiliary building, the staff finds it reasonable that the FLEX Boric Acid Pumps will not be impacted by the loss of heating.

Operation of the TDAFW pump involves steam flow through the turbine and associated piping. The TDAFW pump is located in a temperature-controlled area and is relied upon immediately at the start of ELAP event. The staff finds it reasonable that low outside temperatures would not have an adverse effect on the TDAFW pumps because of their location in a temperature-controlled area, steam flow through the components provides a heat source, and the components are used early in the ELAP event.

The FIP states that the rate of temperature decrease in the vital battery rooms is expected to be about 1 °F/hr after the 4 hour SBO coping time. Based on the rate of temperature decrease and the time frame to align Phase 3 FLEX DGs for supplemental power, the staff finds it reasonable that the battery rooms will not be impacted by the loss of heating.

In its updated FIP [Reference 44], the licensee stated that all the liquid cooling systems for the FLEX diesel generators are treated with anti-freeze and the recommendations for oil viscosity from vendor information for the major FLEX diesel engines was followed. The staff noted that the Unit 1 and Unit 2 Engineering Changes EC-0000053212, "Unit 1 FLEX Mitigation Strategies Overall EC," and EC-0000053213, "Unit 2 FLEX Mitigation Strategies Overall EC," provides the supporting documents from the respective commercial vendors related to the FLEX equipment specifications.

The licensee's core cooling and SFP inventory control FLEX strategies rely on the use of hoses, a FLEX Lift pump and FLEX Booster pump deployed within the protected area; thus, there is a potential for freezing concerns if there is no flow in the hoses in extreme cold weather. During the audit, the staff noted that procedures for operation of the FLEX Lift pump and FLEX Booster pump, 12-OHP-4027-FSG-311 and 12-OHP-4027-FSG-312, respectively, that guidance is provide to operators to drain the pump and hoses to prevent freeze damage when ambient temperatures are below 40 °F. The staff finds this guidance to operators to be reasonable for ensuring FLEX equipment remains capable of supporting FLEX strategies during extreme cold conditions.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

The NRC staff reviewed CNP calculations, MD-12-HV-013-N, "AB & CD Battery Rooms Hydrogen Evolution," Rev. 2, and MD-12-HV-022-N, "N-Train Battery Room Hydrogen Analysis and Maximum Temperature During Normal Plant Operation," Rev. 1, to verify that hydrogen gas accumulation in the 250 Vdc Vital Battery rooms will not reach combustible levels when heating, ventilation, and air conditioning (HVAC) systems are lost during an ELAP. The licensee's analysis considered hydrogen gas generation rates provided by the battery manufacturer (C&D Technologies) during an equalize charge, float charge, and worst-case maximum temperatures (120 °F).

Based on its review of the analysis, the NRC staff concluded that hydrogen accumulation in the 250 Vdc vital battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP since it is assumed that power will be restored to the vital battery rooms HVAC systems and battery room doors opened within the calculated times before hydrogen gas accumulation reaches 2 percent in the 250 Vdc vital battery rooms.

3.9.2 Personnel Habitability

3.9.2.1 Main Control Room

Each unit has a separate control room, located in the auxiliary building. In its FIP the licensee indicated that it qualitatively extrapolated the results of the calculation for control room temperatures during a postulated fire event, which used a GOTHIC model and showed a nominal 117°F maximum resultant control room temperature with temporary fans installed. The license further stated that from another calculation, the SBO bounding heat load in the control room is 125,343 btu/hr, which is much less than the 330,000 btu/hr assumed in the calculation during a postulated fire event. The licensee clarified that this SBO heat load reflects equipment powered by the station batteries. Based on its assessment, the licensee determined that the needed actions to provide adequate control room ventilation during the FLEX Phase 1 and Phase 2 response involved opening cabinet doors to provide vital instrument cooling and the installation of temporary fans. During the audit, the NRC staff noted that control room instrumentation cabinet doors are opened as a 30-minute time-credited-action in its procedure, 1/2-OHP-4023-ECA-0.0, "Loss of All AC Power," for responding to loss of all AC power. Furthermore, the staff noted during the audit that procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," Rev. 1, directs installation of temporary fans in response to a loss of control room ventilation and provides a sketch of the approximate fan locations. Based on the licensee's deployment of temporary portable fans and the expected temperature response in the control room, the staff noted that it is reasonable that the control room will remain habitable during an ELAP event.

3.9.2.2 Spent Fuel Pool Area

There is a single SFP, which serves both units, located in a section of the auxiliary building. During the audit, the NRC staff noted that CNP calculation, PRA-STUDY-095, "Spent Fuel Pool (SFP) Heat Input and Removal Comparison," dated February 22, 2012, indicates that a loss of SFP cooling will result in a heat-up rate of about 0.875 °F/hr and that the licensee indicated that the expected initial SFP temperature is approximately 90 °F. Thus, the licensee established a time constraint of 10 hours from the start of the ELAP event to ensure the SFP area remains habitable for personnel entry for staging FLEX hoses and establishing a vent path. During the audit, the staff noted that procedure, 12-OHP-4027-FSG-11, "Alternate SFP Makeup and Cooling," provides guidance to align a vent pathway for the SFP by opening the roll-up door and SFP roof fire dampers. In addition, the licensee's SFP inventory control strategies are contained in procedure 12-OHP-4027-FSG-11, which provides guidance for restoring SFP level using an alternate source, and procedure 12-OHP-4027-FSG-1101, "Alternate SFP Makeup Equipment Deployment FLEX Lift pump Operation," which provides guidance for deployment of Phase 2 hoses to supply makeup to the SFP via hose to a monitor nozzle or to the SFP demineralizer fill connection. The staff also noted that 12-OHP-4027-FSG-11 provides cautions to operators when performing actions in the area of the SFP regarding habitability and radiation concerns from SFP boiling and reduced SFP levels. The staff noted it is reasonable that the area of the SFP will remain habitable within the timeframe the licensee's procedures direct an operator to deploy hoses to the SFP deck level or fill connection as well as aligning a vent path.

3.9.2.3 Other Plant Areas

There are three FLEX Boric Acid Pumps stored in the auxiliary building at elevation 587 ft. The pumps are mounted on mobile carts and can be moved to the vicinity of the CVCS Reciprocating Charging Pump or SIS rooms at that same building elevation. The FLEX Boric Acid Pumps are powered by a FLEX 250 kW DG with a deployed location outside the auxiliary building crane bay rollup door. During the audit, the licensee identified the deployed location of the FLEX Boric Acid Pumps and the staff noted it is not in a confined area of the auxiliary building, which is temperature controlled prior to the ELAP event. Based on the intermittent use of the FLEX Boric Acid Pumps and its deployed location within the auxiliary building, the staff finds it reasonable that the area of the FLEX Boric Acid Pump will remain habitable during an ELAP Event. Furthermore, the staff noted that the FLEX 250 kW DGs are not deployed within the auxiliary building; thus, there are no concerns regarding diesel exhaust fumes and personnel habitability.

The licensee's Phase1 core cooling FLEX strategies rely on the TDAFW pumps as the motive force for providing water to the SGs. The TDAFW pump trip and throttle valve, electronic governor, and all associated MOVs are powered by 250 Vdc from the N train battery. However, in the unlikely event that that dc power is lost, the trip and throttle valve can be manually opened and turbine overspeed would be prevented by a mechanical governor, and SG feedwater valves can be manually operated. The staff noted that procedure 12-OHP-4027-FSG-501, "FLEX Equipment Staging," directs personnel to open the doors to the affected TDAFW Pump Rooms, which increases the volume of air available to dissipate heat. In addition, the procedure directs personnel to install a temporary portable ventilation fan with an exhaust duct routed out of the room and outside the hallway to the Turbine Building to provide supplemental cooling and a flow path to exhaust heated air out of the TDAFW pump rooms. Based on these licensee procedures for opening up TDAFW pump room doors and setting up portable exhaust fans, the staff finds it reasonable that the TDAFW pump rooms will remain habitable during an ELAP event.

The FIP states that manual control of the SG PORVs is credited for venting steam to remove decay heat, which involves an operator on each unit entering a steam stop enclosure and opening the two PORVs in that enclosure and then exiting the room. The same operation is performed in the opposite steam stop enclosure, then the operators are expected to leave the steam stop enclosures when not actually operating the PORVs and await further direction from the control rooms. During its on-site audit the staff noted that operation of the SG PORVs are part of the licensee's existing procedures in response to an SBO. The licensee stated that the acceptability of the steam stop enclosure environment during an SBO was evaluated in calculation MD-12-MSC-054-N, "Dominant Areas of Concern for Station Blackout Evaluation," which determined a 150 °F expected temperature to be acceptable. The licensee explained that for an ELAP event it used calculation MD-12-MSC-054-N because the SBO conditions bound the ELAP event for the first four hours. The staff noted that the licensee's time validated sequence of events for an ELAP event show that operators typically start the plant cooldown in approximately one hour, and at the direction of the control room supervisors it may be necessary for the operator to re-enter the room to adjust the PORV positions. The licensee anticipates that adjustment of the SG PORVs would take no longer than 10 minutes for each entry and the ELAP response cooldown would be completed within one hour of initiating the cooldown. The staff noted that the expected time frame to complete the ELAP response cooldown within two hours of the event initiation is within the four-hour time frame assessed in MD-12-MSC-054. Furthermore, in its FIP the licensee stated that specialized personal protective equipment, provided as part of the FLEX implementation, will enhance the operator's ability to perform the required function in the expected steam stop enclosure environment. During the audit review, the licensee explained that this protective equipment is heat resistant suits (aluminized carbon/Kevlar suits designed for use by foundry workers). Based on the licensee's time validation for an ELAP initial cooldown being encompassed by calculation MD-12-MSC-054 and the availability of protective equipment to operators, the staff finds it reasonable that operators can accomplish required actions within the steam stop enclosures during an ELAP event.

3.9.3 Conclusions

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

3.10 Water Sources

3.10.1 Steam Generator Make-Up

Phase 1

The licensee stated in its FIP that each unit has one CST, which provides a qualified source of water for the TDAFW pumps to provide water to the SGs for heat removal from the RCS. The license stated that the CSTs are Seismic Class II components qualified to withstand Seismic Class I loads, and are located above the probable maximum flood elevation. During the audit, the NRC staff noted that the licensee performed evaluations (calculation No. 32-9222624-004 and calculation No. 32-9222496-002) that determined the CSTs will survive impact from the design-basis missiles up to a tank height of 16' 7". The licensee indicated that the survival of the CSTs up to a height of 16' 7" ensures that there is sufficient water available for suction to the Unit 1 and 2 TDAFW pumps for at least 12 hours, which allows time for portable Phase 2 FLEX equipment to be deployed. In addition, the licensee evaluated the Unit 1 and 2 CSTs for design-basis wind loading with coincidental atmospheric pressure drop (calculation No. SD-140415-001) to demonstrate that the tank will remain intact and structurally adequate to hold the inventory for an acceptable coping period. The licensee's evaluation determined that buckling is possible near the top 2 feet of the tank; however, this is a localized effect and will not affect the overall integrity of the tank and its ability to retain inventory. The staff noted that the overall tank height is approximately 34' and based on calculation No. 32-9222496-003 the licensee is only crediting the water inventory up to 16'-7" of the CSTs. In its FIP, the licensee identified that there are large bore penetrations on each CST and two nominal 1" penetrations located within one foot of the bottom of each tank that are in the same quadrant as the large bore penetrations. The licensee performed an assessment (calculation No. 51-9222950-002) that determined that intervening structures (e.g., refueling water storage tanks, primary water storage tanks, containment buildings) would provide protection for these penetrations. During the audit, the staff noted that the RWSTs, primary water storage tanks and containment buildings offer shielding for these penetrations of the CSTs, such that a tornado missile would need to travel a tortuous path to strike these penetrations. Thus, the licensee has demonstrated that the CSTs are protected from all applicable hazards and are available to support Phase 1 core cooling strategies. After depletion of the CST inventory, the TDAFW pump suctions will be aligned to the plant ESW supply line, which will be pressurized with water from Lake Michigan using the FLEX Lift pump.

If implemented appropriately and consistent with the FIP, the licensee should have water sources available during the Phase 1 core cooling strategies for SG inventory makeup. In addition, the licensee's strategy should provide sufficient time for operators to deploy and staging Phase 2 FLEX equipment. The licensee's sequence of events timeline, as documented in Table 3 of its FIP, shows that the FLEX Lift pump can be deployed and aligned for service well within the 12 hours provided by available CST inventory.

Phase 2

The licensee stated in its FIP that the Phase 2 FLEX strategy for reactor core cooling and heat removal uses water from Lake Michigan for feeding the SGs using a FLEX Lift pump. The discharge of the FLEX Lift pump feeds either the ESW supply line to the TDAFW pump or feeds a FLEX Booster pump that provides makeup to the SGs through the MDAFW header or the main feedwater header. The staff noted that the supply of water from Lake Michigan to the plant forebay is robust and is protected from all applicable hazards and is available to support Phase 2 core cooling strategies. Per UFSAR Section 2.9.2, the screen house structure (which includes the forebay) is Seismic Class I to the extent needed to provide water to the essential service water pumps. Per UFSAR Section 2.9.5, the Seismic Class I structures were evaluated for tornado conditions to assure that there would be no loss of function.

In addition, the licensee evaluated the impact on the SGs from injecting lake water for core cooling in Phase 2. The evaluations addressed the consequences of using non-standard coolant in the secondary system, including the effects on the integrity of the SG tubes, the steam flow path from the two-phase mixture level in the SG through the TDAFW pump, and the feedwater side flow path up to the SG. This included consideration of the impact of corrosion of the secondary side carbon steel with specific attention to the impact on the integrity of the main steam lines as well as the impact due to phase transitions that result in precipitation or volatilization of chemical species in the SGs and secondary system. The evaluations also determined the potential impact on the SG heat exchanger surface and the potential impact on the components of the TDAFW turbine and pump. The evaluation concluded that use of Lake Michigan water would not compromise system integrity or equipment performance that would preclude maintaining the plant in a safe condition for at least 98 hours when feeding two SGs per unit or 199 hours when feeding four SGs per unit. These times are well in excess of the expected time for transitioning to RHR cooling in Phase 3.

In its FIP, the licensee addressed frazil ice formation on Lake Michigan and its potential impact on accessing the UHS as follows:

> Frazil ice is a surface and sub-surface phenomena associated with large bodies of water under extremely cold, windy and turbulent conditions. This results in emulsified ice crystals in the surface and subsurface of the large water body. The CNP intake structure and forebay connect to Lake Michigan through three large intake tunnels with intake cribs mounted on the lake bottom. The intake cribs are significantly below the lake surface. During an ELAP event, the CW, ESW and NESW [non-essential service water] pumps will all be stopped. Under the drastically reduced flow rates considered during an ELAP, the forebay acts as a stilling well. Communication with Lake Michigan maintains the water supply while shielding the forebay from the adverse effects of surface freezing and frazil ice production. Based on the construct and design of the lake intake structures and CW system forebay, reasonable assurance exists that induction of frazil ice will not be a concern.

> Additionally, with lake water temperatures in the range required for frazil ice formations, one of the unit's CW systems would normally be aligned in the de-ice mode of operation. Therefore the water temperature in the forebay would be greater than 32 °F at the start of the BDBEE. Since frazil ice formation requires supercooled liquid, the initial BDBEE conditions would inhibit the formation of frazil ice.

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

Phase 3

The licensee indicated in its FIP that Lake Michigan would be utilized as the heat sink during Phase 3 reactor core cooling and heat removal. Two large capacity raw water pumps will be provided from the NSRC and deployed to the intake structure to provide Lake Michigan water to the ESW system. Each large capacity pump will be fed by two floating lift pumps. The ESW system provides water to cool the CCW heat exchangers to remove heat from the RHR system. Restoring RHR cooling will provide for indefinite core cooling capability. The CCW and RHR pumps will be powered by the four NSRC-supplied 1 MW, 4.16 kV diesel-fueled turbine-driven generators.

If implemented appropriately and consistent with the FIP, the licensee should have an adequate source of water available during the Phase 3 core cooling.

3.10.2 Reactor Coolant System Make-Up

Phase 1

In its FIP, the licensee stated that no actions would be needed in Phase 1. Westinghouse Generation 3 SHIELD® low leakage seals were installed in the Unit 1 and Unit 2 RCPs during the fall 2014 refueling outage and the spring 2015 refueling outage, respectively. The low leakage seals limit the total RCS leak rate to no more than 5 gpm (1 gpm per RCP seal and 1 gpm of unidentified RCS leakage). With credit for the SHIELD® seals and the passive injection of accumulator inventory, Westinghouse analyses demonstrate that natural circulation in the RCS could be maintained for multiple days under postulated ELAP conditions without reliance upon FLEX RCS injection. However, the licensee conservatively determined that RCS boration should be initiated by 16 hours into the event to ensure adequate shutdown margin. Thus, providing borated RCS makeup within 16 hours would avoid inadvertent recriticality and conservatively prevent the transition to reflux cooling. Therefore, the Phase 1 strategy for ensuring adequate RCS inventory and reactivity control requires no external sources of water.

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

Phase 2

In its FIP, the licensee stated that the BASTs would be the primary source for RCS makeup and boration to compensate for contraction of the RCS coolant due to cooldown and for RCS leakage, and to provide reactivity control in Phase 2. The FLEX boric acid pumps would take suction from a hose connected to the boric acid transfer pump suction header, and discharge to the RCS through a tee connection installed on the charging pumps discharge header. The licensee stated that one BAST contains sufficient volume to maintain one reactor core subcritical following the cooldown to just below 200 °F.

There are three BASTs. The tanks are constructed of stainless steel and are at atmospheric pressure. Each BAST has a capacity of 11,000 gallons. The BASTs are located within the auxiliary building and are protected from external hazards. The BASTs are classified as Seismic Class I. Per the CNP Technical Requirements Manual, TRM 8.1.1, in Modes 1 and 2 one BAST must be operable for each unit, with at least 8,500 gallons of water with ≥ 6,550 ppm boron concentration in an operable BAST.

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

Phase 3

The licensee stated that the need for RCS makeup will be reduced when in the RHR cooling mode for decay heat removal. In procedure FSG-1, "Long Term RCS Inventory Control," if the BASTs and the boric acid reserve tank (BART) are not available, then the RWSTs are used. There is one RWST for each unit. The RWSTs each have a nominal capacity of 420,000 gallons of borated water, and are required by Technical Specifications to have at least 375,500 gallons with 2400 ppm boron in Modes 1 to 4. The tanks are stainless steel, are vented to atmosphere, and are insulated. UFSAR, Section 2.9.2, states that the RWSTs are Seismic Class I structures, designed to withstand the design basis earthquake. CNP calculation 51-9225061, "Donald C. Cook CST and RWST Survivability Report," Appendix B, addressed the survivability of the RWSTs for tornado missiles, and concluded based on their similarity to the CSTs they had a roughly equivalent response as the CSTs. As shown on the drawings of the respective tanks, the CSTs and RWSTs are fabricated from the same material type and have the same weld size and type. Thus, the staff noted it is reasonable that the conclusions from the missile impact assessments performed for the CSTs are also applicable to the RWSTs. Even if a missile were to penetrate the upper section of the tank, the licensee stated that each RWST should retain the credited volume of 232,000 gallons of water. In the FIP, the licensee describes that there are narrow pathways where tornado missiles might impact an RWST nozzle, and for this reason only took credit for one of the RWSTs in an ELAP event. One RWST with the credited volume of 232,000 gallons is sufficient to supply makeup water to the RCS of both units. The licensee relies on the piping from the RWST to the charging pump suction header to bring the RWST water to that header, and then uses a temporary hose to the suction of either of the FLEX boric acid pumps, which are located in the auxiliary building.

If implemented appropriately and consistent with the FIP, the licensee should have a sufficient source of water available during Phase 3 to maintain RCS inventory while cooling down and maintaining RHR cooling.

3.10.3 Spent Fuel Pool Make-Up

There is one SFP that serves both units. The licensee stated in its FIP that water from Lake Michigan will be used for makeup to the SFP using the FLEX lift pump. Approximately 49 hours would be required to boil off the SFP water to a level requiring cooling or the addition of makeup to preclude fuel damage, conservatively assuming a dual unit, full core offload. Due to the initial pool water inventory (23 feet over the top of irradiated fuel assemblies) and boil off rate, no makeup is required in Phase 1. In Phase 2 the FLEX lift pump will provide water to the SFP using a hose and monitor nozzle or direct injection to the SFP. Makeup in this manner would continue indefinitely in Phase 3.

If implemented appropriately and consistent with the FIP, the licensee should have adequate water sources available during Phases 1, 2, and 3 to maintain water level in the SFP.

3.10.4 Containment Cooling

The licensee stated in its FIP that in Phases 1, 2 or 3, no external source of water is needed for maintaining containment pressure or temperature below the design limits.

3.10.5 Conclusions

Based on the information 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 adequately addresses 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 pumps to provide the water initially needed for decay heat removal. If the plant has been shut down and all or most of the fuel has been removed from the RPV and placed in the SFP, there may be a shorter timeline to implement the makeup of water to the SFP. However, this is balanced by the fact that if immediate cooling is not required for the fuel in the reactor vessel, the operators can concentrate on providing makeup to the SFP. The licensee's analysis shows that following a full core offload to the SFP from both units, about 49 hours would be required to boil off the SFP water to a level requiring cooling or the addition of makeup to preclude fuel damage 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 steam-powered pump and allow operators to release steam from the steam generators (which typically occurs when the RCS has been cooled below about 300 °F), another strategy must be used for decay heat removal. On September 18, 2013, NEI submitted to the NRC a position paper entitled “Shutdown/Refueling Modes” [Reference 29], which described methods to ensure plant safety in those shutdown modes. By letter dated September 30, 2013 [Reference 30], 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 meeting the requirements of the order to be able to implement appropriate strategies in all modes. This is done by incorporating FLEX equipment in the existing plant process to manage safety functions when in shutdown modes. Considerations in the shutdown safety 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. On June 16, 2015, the licensee submitted a compliance letter for CNP [Reference 18], which stated that CNP will abide by the NEI position paper entitled "Shutdown / Refueling Modes". In its FIP, the licensee stated that CNP procedure PMP-4100-SDR-001, which controls the management and assessment of shutdown safety, has been revised to be consistent with the recommendations provided in the NEI position paper.

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

3.12 Procedures and Training

Procedures

The licensee stated in its FIP that the inability to predict actual plant conditions that require the use of FLEX equipment makes it infeasible to provide specific procedural guidance for each potential condition and hazard. Considering this, the FLEX Support Guidelines (FSGs) provide guidance that can be employed for a variety of conditions. Clear criteria for entry into FSGs ensures that FLEX strategies are used only as directed for BDBEEs, and are not used inappropriately in lieu of existing procedures. Where FLEX strategies supplement emergency operating procedures (EOPs), abnormal operating procedures (AOPs), severe accident management guidelines, extensive damage mitigation guidelines or other procedures, the applicable procedures or guidelines direct entry into and exit from the appropriate FSG.

The licensee stated in its FIP that FSGs have been developed in accordance with PWROG guidelines. The FSGs provide instructions for implementing available, pre-planned FLEX strategies to accomplish specific tasks in the EOPs or AOPs. The FSGs are used to supplement (not replace) the existing procedure structure that establishes command and control for the event. In its FIP, the licensee states that procedural interfaces have been incorporated into the existing SBO procedure, 1/2-OHP-4023-ECA-0.0, as necessary, to provide appropriate reference to the FSGs, and provide command and control for responding to an ELAP.

Additionally, procedural interfaces have been incorporated into the following procedures to include appropriate reference to FSGs:

- 1/2-OHP-4022-001-005, "Loss of Offsite Power with Reactor Shutdown"
- 1/2-OHP-4022-017-001, “Loss of RHR Cooling”
- 12-OHP-4022-018-001, "Loss of Spent Fuel Pit Cooling"

Initial FSG validation was performed using the NEI FLEX validation process. The FSG maintenance and validation will be performed in accordance with plant procedure PMP- 4027-FSG-001.

Training

The licensee stated in its FIP that document TPD-600-FLEX, "FLEX Training Program Description," describes the BDBEE response training program. The program includes identification of required training for Operations, Maintenance, Engineering and emergency response organization (ERO) personnel. These program lessons were developed and/or revised to assure that personnel proficiency in the mitigation of BDBEEs is adequate and maintained. These programs and controls were developed and have been implemented in accordance with the Systematic Approach to Training process.

The licensee stated in its FIP that for key ERO personnel, initial training has been provided and periodic training will be conducted on BDBEE response strategies and implementing guidelines. Personnel assigned to direct the execution of mitigation strategies for BDBEE responses 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) to operator training for BDBEE accident mitigation. The testing and evaluation of operator knowledge and skills in this area have been similarly weighted. Operator training includes training on use of equipment from the NSRC where applicable.

The licensee stated in its FIP that CNP simulator models have been upgraded to provide for Operations Control Room crews to train on implementation of FLEX strategies. The simulator modeling allows training on a BDBEE occurring with the unit at power or shutdown.

Conclusions

The NRC staff finds that the licensee has adequately addressed the procedures and training associated with FLEX. The procedures have been issued and a training program has been established and will be maintained in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 [Reference 31], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 32], the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs. Preventative maintenance templates for the major FLEX equipment including the portable diesel pumps and generators have also been issued.

In its FIP, the licensee stated that PMP-4027-FSG-002, "FLEX Equipment Program," provides the guidance to ensure that Phase 2 FLEX equipment is maintained per the guidance provided in NEI 12-06. The program describes:

- Ownership and responsibilities,
- Planned and unplanned unavailability tracking and source data,
- Preventive maintenance and testing of equipment.

The FLEX equipment program ensures the equipment is maintained to the standards of NEI 12-06 Section 11.5, which endorses the Institute of Nuclear Power Operations (INPO) guidance found in INPO AP 913, "Equipment Reliability Process," and the EPRI associated bases to define site specific maintenance and testing. The planned preventative maintenance activities for cables include inspection and testing which follows the industry recommended practices.

In Enclosure 3 of its compliance letter [Reference 18], the licensee stated that the SAFER equipment is maintained under the PEICo maintenance, testing, and calibration program. SAFER has developed equipment maintenance instructions in accordance with the EPRI templates for the Phase 3, vendor recommendations, and the pooled inventory management rules and procedures (PRPs), which require equipment in the program to be maintained in a serviceable, deployable condition.

Conclusions

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

3.14 Alternatives to NEI 12-06, Revision 0

3.14.1 Reduced Set of Hoses and Cables As Backup Equipment

In its updated FIP, the licensee stated they will use the alternative to NEI 12-06 which permits the use of a reduced set of cables and hoses as the backup (N+1) equipment. The NRC endorsement of that alternative was given in a letter dated May 18, 2015 [Reference 43]. The licensee stated that they will use Method 1 from the endorsement letter.

The NRC staff finds that although the guidance of NEI 12-06 has not been met, if this alternative is implemented as described in the NRC endorsement letter, the licensee will adequately address the requirements of Order EA-12-049 associated with diverse methods of protection.

3.14.2 Spray Flow to the SFP

The NRC staff determined that the licensee has the three methods for SFP makeup stated in NEI 12-06, Table D-3, with the capability for a flow rate exceeding the boil-off rate based on a conservative plant-specific analysis of the fuel's decay heat. However, the staff finds that the licensee's capability for spray flow to the SFP does not fully meet the intent of NEI 12-06, due to the timing of when spray flow can be supplied to the SFP. Therefore, the licensee's approach is an alternative to NEI 12-06. This alternative has been accepted by the NRC staff as stated in section 3.3.4.3 of this safety evaluation.

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, will adequately address the requirements of Order EA-12-049.

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 27, 2013 [Reference 34], the licensee submitted an OIP for CNP in response to Order EA-12-051. By letter dated June 19, 2013 [Reference 35], 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 36]. By letter dated November 13, 2013 [Reference 37], the NRC staff issued an ISE and RAI to the licensee. By letter dated August 14, 2014 [Reference 17] the NRC issued an audit report on the licensee's progress.

By letters dated August 26, 2013 [Reference 38], February 27, 2014 [Reference 39], and August 27, 2014 [Reference 40], the licensee submitted status reports for the Integrated Plan. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable SFP level instrumentation, which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated December 16, 2014 [Reference 41], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has one SFP which serves both units. The licensee has installed a SFP level instrumentation system designed by Mohr Test and Measurement, LLC. The NRC staff reviewed the vendor's SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports. The staff issued a vendor audit report on August 27, 2014 [Reference 42]. Refer to section 2.2 above for the regulatory background for this section.

4.1 Levels of Required Monitoring

Attachment 2 of Order EA-12-051 states in part:

> All licensees identified in Attachment 1 to this 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 [Level 1], (2) level that is adequate to provide substantial radiation shielding for a person standing on the SFP operating deck [Level 2], and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred [Level 3].

By letter dated July 11, 2013, the licensee provided clarification on the proposed Level 1. The licensee stated that the correct elevation associated with Level 1 is 645 ft.1 ½ in. and provided justification of cooling system operation. In a letter dated February 27, 2014, the licensee stated that the elevation associated with Level 2 is 630 ft. 10 ½ in., which corresponds to approximately 10 ft. above the top of the SFP fuel storage rack. The licensee designated Level 3 as an elevation of 620 ft. 10 ½ in., which is the highest point of any spent fuel storage rack seated in the SFP. By letter dated July 11, 2013, the licensee provided a figure of the SFP with the approximate locations identified as Levels 1, 2 and 3 consistent with the licensee's proposed elevations. The NRC staff confirmed the elevations during the CNP on-site audit.

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

4.2 Evaluation of Design Features

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

4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the primary and backup SFP channels will consist of fixed components. In its letter dated February 27, 2014, the licensee clarified that the instrument is capable of measuring water level over a 27 ft. 10 ½ in. range from 649 ft. 1 in. to 621 ft. 2 ½ in. Level 1 is bounded at elevation 645 ft. 1 ½ in., and Level 3 is within +/- 1 ft. at elevation 620 ft. 10 ½ in., as allowed in the guidance.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed design, with respect to the number of channels and measurement range for its SFP, is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.2 Design Features: Arrangement

In its letter dated August 26, 2013, the licensee provided sketches showing the location of the level sensors and the proposed routing of the cables that will extend from the sensors to the location of the readout/display device. These drawings show the location of the level sensors, which would be installed in the northwest and northeast corners of the SFP. In addition, the drawings showed the proposed cable routing from the SFP to the control rooms for Units 1 and 2. In particular, the instrument channel for the probe in the northwest corner of the SFP would be routed to the display to be installed in the Unit 1 control room; and the instrument channel for the probe in the northeast corner of the SFP would be routed to the display to be installed in the Unit 2 control room.

During the onsite audit, the NRC staff walked down the SFP area including the proposed SFPI locations and the primary and back-up cable routes which were partially installed. The NRC staff also observed the proposed display locations in the Unit 1 and Unit 2 control rooms.

The NRC staff noted that there is sufficient channel separation within the SFP area between the primary and back-up level instruments, 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 SFPI is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.3 Design Features: Mounting

During the week of May 26, 2014, the NRC staff conducted an audit of the design and qualification of Mohr's SFPI including site-specific analysis and testing for CNP. The staff issued an audit report dated August 27, 2014 [Reference 42].

Seismic test results for the SFPI signal processing unit and the extended battery are documented in Mohr's document No. 1-4010-6, "Seismic Test Report" Rev. 1, dated February 6, 2014. These tests were conducted on a triaxial shake table using the Institute of Electrical and Electronics Engineers (IEEE) guidance of IEEE 344- 2004, Sections 7, 8, 9, and 10 as recommended in the NRC's document JLD-ISG-2012-03 [Reference 9].

The NRC staff reviewed Mohr's document No. 1-041 0-9.1, "SFP-1 Site-Specific Seismic Analysis Report: Donald C. Cook Nuclear Plant (D.C. Cook)," Rev.1, dated May 1, 2014. This seismic test report is specific to the CNP site for the signal processing unit and the extended battery enclosure. The results of the analysis show that the calculated stresses are well within the allowable stresses. This site-specific analysis conservatively assumed that the CNP probe is 10 in. from both the SFP walls in the selected corners of the SFP and will not impact the pool liner during the postulated seismic event and resulting sloshing.

Mohr's document No. 1-0410-9, "SFP-1 Level Probe Assembly Seismic Analysis Report," Rev. 2, dated May 12, 2014, addresses the seismic adequacy of the SFP level probe assembly. This report is generic in nature and each licensee should verify that its site-specific analysis for their probe assembly is enveloped by this test's parameters. In this instance, the seismic test criteria used in the report are significantly higher than the seismic criteria for CNP. Seismic loads and SFP water sloshing loads were included in this report. The sloshing analysis was based on GOTHIC, an industry-standard computer code for performing multiphase fluid flow. ANSYS, a finite element analysis computer code, was used to perform the hydrodynamic loading and structural analysis. A series of sensitivity analyses were conducted, which included pool length, boundary conditions, liquid height and loss of SFP inventory. The report indicates that the distance from the probe to the pool liner is 12 in. for the analyses performed. This distance may vary depending on site-specific installation. The analysis concluded that the maximum stresses on the probe are lower than the maximum allowable stresses.

Calculations performed during the generic testing at levels above the CNP site-specific seismic criteria showed that the probe has a high likelihood of impacting the SFP metal liner several times during a seismic event. To verify the probe's functionality under such conditions, Mohr performed physical impact tests and test results confirmed that mechanical integrity of the probe and the SFP liner were preserved. The pictures after the tests showed little to no damage to the liner or the probe.

Analysis was also performed to derive shear forces, axial loadings, and bending moments at the mounting flange location due to seismic and hydrodynamic loading of the probe. Test results are documented in Table 1 of Mohr's document No. 1-0410-9 and include flange forces, bending moments, peak displacement, and peak probe velocity in all three directions (x, y, and z directions). All the stress forces are within the allowable limits. Based on the test results from this generic analysis, Mohr found the SFPI level probe assembly acceptable for use.

During the CNP onsite audit, the staff walked down the proposed location for the mounting brackets on the north side of the SFP.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed mounting design is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses 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 [Reference 34], the licensee stated that instrument channel reliability shall be established by use of an augmented quality assurance process similar to that described in NEI 12-02.

If implemented appropriately, this approach is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.4.2 Instrument Channel Reliability

NEI 12-02 states:

> The instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and/or testing of channel components for the following sets of parameters, as described in the paragraphs below:

- conditions in the area of instrument channel component use for all instrument components,
- effects of shock and vibration on instrument channel components used during any applicable event for only installed components, and
- seismic effects on instrument channel components used during and following a potential seismic event for only installed components.

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

During the vendor audit for Mohr Test and Measurement, LLC, the NRC staff reviewed vendor documents associated with testing conducted to demonstrate equipment reliability under BDB conditions. The staff reviewed the test results related to environmental qualification for the coaxial cable, probe assembly, and mounting flange located inside the SFP area and outside where the electronics and battery enclosures will be located. The staff included a summary of the SFPI environmental qualification and reliability design documents reviewed in the audit report dated August 27, 2014.

The NRC staff notes that the SFPI environmental specifications, as designed by the vendor, meet or exceed the environmental conditions at CNP for the SFPI installed locations for all the instrument components.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed instrument qualification process is consistent with NEI 12-02 guidance, as endorsed, by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.5 Design Features: Independence

In its February 27, 2014, letter, the licensee stated that the primary and back-up SFPI channels are completely independent and that sensor probes are installed on opposite corners of the north side of the SFP to maintain adequate channel separation. During the onsite audit visit, the NRC staff walked down the SFP area and the route for the primary and back-up cables. Additionally, each instrument channel is normally powered by 120 Vac distribution panels powered by different 480 Vac buses to support continuous monitoring of the SFP level.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed design, with respect to instrument channel independence, is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.6 Design Features: Power Supplies

In its letter dated February 27, 2014, the licensee stated that each SFPI instrument channel is normally powered by a different 120 Vac distribution panel. Each of those 120 Vac panels is powered by a different 600 Vac bus. On loss of normal 120 Vac power, each channel is equipped with a separate uninterruptible power supply (UPS) that will automatically transfer to a dedicated 72-hour back-up battery. If normal power is restored, then the instrument channels will automatically transfer back to the normal 120 Vac power source. Mohr Test and Measurement, LLC, document 1-0410-7, "MOHR EFP-IL Spent Fuel Pool Instrumentation System Battery Life Report," concluded that the batteries have sufficient capacity to maintain the level indication function for longer than 7 days. Mohr Document No. 1-0410-10, "MOHR EFP-IL SFPI System Power Interruption Report," Rev. 1 dated January 10, 2014, describes power interruption testing on the EFP-IL signal processing unit and battery. Test results indicate that no deficiencies were identified with respect to maintenance of reliable function, accuracy, or calibration as a result of power interruption.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed power supply design is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.2.7 Design Features: Accuracy

Mohr document No. 1-0410-3, "MOHR EFP-IL SFPI Proof of Concept Report," Rev. 0, dated October 17, 2012, states, in part, that the effects of temperature and humidity are insignificant with regard to measurement accuracy. The instrument accuracy is expected to be 0.5 inch based on the results from testing performed on the probe at 500 °F in saturated steam (100 percent relative humidity). Mohr Document No. 1-0410-15, "MOHR EFP-IL-SFPI System Uncertainty Analysis," states, in part, that the EFP-IL-SFPI system, configured with a maximum length of transmission cable of 1000 ft., stays within the level measurement accuracy of +/- 3 in. The EFP-IL-SFPI system error is highest, but still acceptable, at the bottom of the probe near the top of the fuel rack.

Mohr Document No. 1-0410-10, "MOHR EFP-IL SFPI System Power Interruption Report," Rev. 1 dated January 10, 2014, describes power interruption testing on the EFP-IL signal processing unit and battery. Test results indicate that no deficiencies were identified with respect to maintenance of reliable function, accuracy, or calibration as a result of power interruption.

The NRC staff included a summary of the SFPI accuracy evaluation and documents reviewed in the audit report dated August 27, 2014.

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

4.2.8 Design Features: Testing

During the week of May 26, 2014, the NRC staff audited a number of vendor documents related to testing. Mohr document No. 1-0410-12, "MOHR EFP-IL Signal Processor Operator's Manual," 1-0410-13, "MOHR EFP-IL Signal Processor Technical Manual," and 1-0410-14, "MOHR SFP-1 Level Probe Assembly Technical Manual" provide the testing and calibration procedures for the SFPI.

Mohr's SFPI design can be calibrated in-situ without removal from its installed location. The system is calibrated using a CT-100 device and processing of vendor scanned files. Mohr document No. 1-0410-13, Section 6.5, "Calibration," provides recommended calibration intervals to be followed by users of this technology. The NRC staff was informed that CNP personnel have visited the vendor site and received training on maintenance, installation and operation of Mohr's SFPI technology.

The NRC staff included a summary of the SFPI testing documents reviewed in the audit report dated August 27, 2014.

In its letter dated February 27, 2014, the licensee stated that functional checks are automated and/or semi-automated (requiring limited operator or technician interaction) and are performed through the instrument menu software and initiated by the operator or technician. There are a number of other internal system tests that are performed by system software on an essentially continuous basis without user intervention but which can also be performed on an on-demand basis with diagnostic output to the display for the operator or technician to review. Other tests such as menu button tests, level alarm, and alarm relay tests are only initiated manually by the operator or technician. At a minimum, functional checks will be performed at a frequency commensurate with vendor recommendations.

Calibration checks are described in detail in the Vendor Operator's Manual, and the licensee stated that applicable information will be contained in plant procedures or preventive maintenance tasks. At a minimum, calibration checks will be performed at a frequency commensurate with vendor recommendations, not to exceed the calibration frequency required by Order EA-12-051.

Channel checks, or checks of one independent channel against another, can be conducted because each instrument electronically logs a record of measurement values over time in non-volatile memory that can be compared to demonstrate constancy, including any changes in pool level, such as those associated with the normal evaporative loss/refilling cycle. The channel level measurements can be directly compared to each other (i.e., regular cross-channel comparisons). Direct measurements of SFP level may be used for diagnostic purposes if cross-channel comparisons are anomalous.

Formal calibration checks are recommended by the vendor on a two-year interval to demonstrate calibration to external National Institute of Standards and Technology-traceable standards. Formal calibration check surveillance interval and timing will be established consistent with the requirements of Order EA-12-051.

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

4.2.9 Design Features: Display

Section 3.9 of NEI 12-02 states that the intent of the guidance is to ensure that information is promptly available to the plant staff and decision makers. Ideally, there will be an indication from at least one channel of instrumentation in the control room.

In its letter dated August 26, 2013, the licensee stated that one instrument channel display would be located in the Unit 1 control room, and the second channel display would be located in the Unit 2 control room. In addition, the licensee stated that the channels function in an identical manner, and both are suitable for a primary or backup function.

The NRC staff notes that the NEI guidance for "Display" specifically mentions the control room as an acceptable location for SFP instrumentation displays, as it is occupied or promptly accessible, outside the area surrounding the SFP, inside a structure providing protection against adverse weather, and outside of any very high radiation areas or locked high radiation areas during normal operation.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed location and design of the SFP instrumentation displays is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses 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

NEI 12-02 specifically addresses the use of Systematic Approach to Training (SAT) for training personnel in the use and the provision of alternate power to the primary and backup SFP instrument channels. In its OIP, the licensee indicated that the SAT will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training.

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

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

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed plan to train personnel in the use and the provision of alternate power to the primary and backup instrument channels, including the approach to identify the population to be trained, is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.3.2 Programmatic Controls: Procedures

By letter dated February 27, 2014 [Reference 39], the licensee stated that site specific procedures would be developed for system inspection, calibration and testing, maintenance, repair, operation, and normal and abnormal responses in accordance with CNP procedural controls and will be based on recommended operation and maintenance procedures provided by Mohr. The licensee also provided a list of the procedures that will govern the use of the SFPI. The list included procedures for system inspection, calibration and testing, maintenance, repair, operation, and FSGs. Additionally, the licensee stated that preventive maintenance procedures to include tests, inspection and periodic replacement of the backup batteries will be developed based on Mohr Test and Measurement, LLC's recommendations.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's procedure development is consistent with NEI 12-02 guidance, as endorsed, by JLD-ISG-2012-03, and adequately address the requirements of the order.

4.3.3 Programmatic Controls: Testing and Calibration

In its letter dated February 27, 2014 [Reference 39], the licensee indicated that the periodic calibration verification will be performed consistent with the guidance provided in NEI 12-02 Section 4.3. Provisions associated with out of service (OOS) or non-functional equipment including allowed outage times and compensatory actions will also be consistent with the guidance provided in Section 4.3 of NEI 12-02. If one OOS channel cannot be restored to service within 90 days, appropriate compensatory actions, including the use of alternate suitable equipment, will be taken. If both channels become OOS, actions would be initiated within 24 hours to restore one of the channels to operable status and to implement appropriate compensatory actions, including the use of alternate suitable equipment and/or supplemental personnel, within 72 hours. The licensee has documented these actions in its Technical Requirements Manual. The licensee stated that CNP will maintain sufficient spare parts for the SFPI, taking into account the lead time and availability of spare parts, in order to expedite maintenance activities, when necessary, to provide assurance that both channels are not out of service for an extended period of time.

Based on the discussion above, the NRC staff finds that, if implemented appropriately, the licensee's proposed testing and calibration plan is consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and adequately addresses the requirements of the order.

4.4 Conclusions for Order EA-12-051

In its letter dated February 27, 2013 [Reference 34], the licensee stated that it 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's plans conform to the guidelines of 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 CNP according to the licensee's proposed design, it should adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on these two orders. The staff conducted an onsite audit in June 2014. The licensee reached its final compliance date on April 25, 2015, and has declared that both of the CNP 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 and which NRC staff has evaluated to be satisfactory for compliance with these orders. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs which if implemented appropriately will adequately address the requirements of Orders EA-12-049 and EA-12-051.

6.0 REFERENCES

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

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

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

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

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

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

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

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

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

10. Donald C. Cook Nuclear Plant, Units 1 & 2 Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049), dated February 27, 2013 (ADAMS Accession No. ML13101A381)

11. First Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-

Design-Basis External Events dated August 26, 2013 (ADAMS Accession No. ML13240A308)

12. Second Six Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events dated February 27, 2014 (ADAMS Accession No. ML14063A042)

13. Third Six Month Status Report for the Implementation of Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated August 27, 2014 (ADAMS Accession No. ML14241A235)

14. D.C. Cook, Unit 2 - Fourth Six Month Status Report for the Implementation of Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated February 25, 2015 (ADAMS Accession No. ML15058A032)

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 Lawrence J. Weber dated January 24, 2014 regarding Donald C. Cook Nuclear Plant, Units 1 and 2 – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Order EA-12-049 (Mitigating Strategies) (ADAMS Accession No. ML13337A325)

17. Letter from John Boska (NRC) to Lawrence J. Weber dated August 13, 2014 regarding Donald C. Cook Nuclear Plant, Units 1 and 2 – Report for the Onsite Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Instrumentation Related to Orders EA-12-049 and EA-12-051 (ADAMS Accession No. ML14209A122)

18. Donald C. Cook Nuclear Plant Units 1 and 2, Compliance With March 12, 2012, NRC Order Regarding Mitigating Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049), dated June 16, 2015, (ADAMS Accession No. ML15169A107)

19. Donald C. Cook Nuclear Plant Units 1 and 2 - Final Integrated Plan Regarding March 12, 2012 U.S. Nuclear Regulatory Commission Order Regarding Mitigating Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049), dated June 16, 2015, (ADAMS Accession No. ML15169A106)

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

21. 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)

22. Donald C. Cook Nuclear Plant Unit 1 and 2 - Response to March 12, 2012, Request for Information, Enclosure 2, "Recommendation 2.1: Flooding," Required Response 2, Hazard Reevaluation Report, dated March 6, 2015 (ADAMS Accession No. ML15069A334)

23. Donald C. Cook Nuclear Plant Units 1 and 2, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights From the Fukushima Dai-ichi Accident, dated March 27, 2014 and March 17, 2014 (ADAMS Accession Nos. ML14092A329 and ML14092A330, respectively)

24. Letter from Frankie Vega (NRC) to Lawrence J. Weber, Donald C. Cook Nuclear Plant, 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 (TAC Nos. MF 3873 and MF 3874), dated April 21, 2015 (ADAMS Accession No. ML15097A196)

25. 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)

26. 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)

27. 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," dated May 7, 2013 (ADAMS Accession No. ML13106A331)

28. 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)

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

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

31. Donald C. Cook Nuclear Plant Units 1 and 2, Expedited Seismic Evaluation Process Report- Response to U.S. Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated December 18, 2014 (ADAMS Accession No. ML14357A053)

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

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

34. Donald C. Cook Nuclear Plant Unit 1 and Unit 2, Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051), dated February 27, 2013 (ADAMS Accession No. ML13071A323)

35. Letter from Thomas Wengert (NRC) to Lawrence J. Weber dated June 19, 2013 regarding Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information on the Overall Integrated Plan in Response to Order EA-12-051 Concerning Reliable Spent Fuel Instrumentation (ADAMS Accession No. ML13164A381)

36. Donald C. Cook Nuclear Plant Units 1 and 2, Response to Request for Additional Information Regarding the Overall Integrated Plan in Response to Order EA-12-051, Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated July 11, 2013 (ADAMS Accession No. ML13196A250)

37. Letter from Thomas Wengert (NRC) to Lawrence J. Weber, regarding Donald C. Cook Nuclear Plant, Units 1 and 2 – Interim Staff Evaluation and Request for Additional Information Regarding the Overall Integrated Plan For Implementation of Order EA-12-051, Reliable Spent Fuel Instrumentation (ADAMS Accession No. ML13310B499)

38. Donald C. Cook Nuclear Plant Units 1 and 2, Six Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051), dated August 26, 2013 (ADAMS Accession No. ML13247A050)

39. Donald C. Cook Nuclear Plant Units 1 and 2, Six Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051), dated February 27, 2014 (ADAMS Accession No. ML14063A041)

40. D.C. Cook, Units 1 and 2, Third Six-Month Status Report in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051), dated August 27, 2014 (ADAMS Accession No. ML14241A236)

41. Donald C. Cook, Units 1 and 2, Compliance With March 12, 2012, U.S. Nuclear Regulatory Commission (NRC) Order Regarding Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051), dated December 16, 2014, (ADAMS Accession No. ML14352A231)

42. Letter from John Boska (NRC) to Lawrence J. Weber dated August 27, 2014 regarding Donald C. Cook Nuclear Plant, Units 1 and 2 – Report for the Onsite Audit of Mohr Regarding Implementation of Reliable Spent Fuel Instrumentation Related to Order EA-12-051 (ADAMS Accession No. ML14216A362)

43. Letter from Jack R. Davis (NRC) to Joseph E. Pollock (NEI), regarding NRC endorsement of an alternative to NEI 12-06 regarding the N+1 equipment for hoses and cables, dated May 18, 2015 (ADAMS Accession No. ML15125A442)

44. Donald C. Cook Nuclear Plant Units 1 and 2 – Revision 1 of Final Integrated Plan Regarding March 12, 2012 U.S. Nuclear Regulatory Commission Order Regarding Mitigating Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049), dated October 1, 2015 (ADAMS Accession No. ML15280A023).

45. 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," dated September 1, 2015 (ADAMS Accession No. ML15174A257).

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

47. Westinghouse Report WCAP-17601-P, Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs, Revision 1, dated January 31, 2013 (Proprietary).

48. Letter from Jack R. Davis (NRC) to Jack Stringfellow (PWROG), regarding NRC endorsement of Westinghouse Position Paper: "Westinghouse Response to NRC Generic Request for Additional Information (RAI) on Boron Mixing in Support of the Pressurized Water Reactor Owners Group (PWROG)", dated January 8, 2014 (ADAMS Accession No. ML13276A183).

49. Pressurized-Water Reactor Owners Group Document PWROG-14064-P, Application of NOTRUMP Code Results for Westinghouse Designed PWRs in Extended Loss of AC Power Circumstances, Revision 0, dated September 30, 2014, (ADAMS Accession No. ML14276A100-Proprietary)

Principal Contributors: L. Okruhlik
J. Miller
J. Lehning
K. Scales
B. Titus

O. Yee
S. Wyman
J. Boska
R. Mychajliw

Date: November 9, 2015

the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated November 13, 2013 (ADAMS Accession No. ML13310B499), and August 13, 2014 (ADAMS Accession No. ML14209A122), the NRC staff issued an ISE and audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated December 16, 2014 (ADAMS Accession No. ML14352A231), I&M 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 I&M's strategies for CNP. The intent of the safety evaluation is to inform I&M on whether or not its integrated plans, if implemented as described, provide a reasonable path for compliance with Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 191, "Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/ Multi-Unit Dose Assessment Plans" (ADAMS Accession No. 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, CNP Project Manager, at 301-415-2901 or at John.Boska@nrc.gov.

Sincerely,

/RA/

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

Docket Nos.: 50-315 and 50-316
Enclosure:
Safety Evaluation
cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC
JLD R/F
RidsNrrDorlLp3-1 Resource
RidsNrrLASLent Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn3MailCenter Resource
JBoska, NRR/JLD
RidsNrrPMDCCook Resource
LGibson, NRR/JLD

ADAMS Accession No. ML15264A851 ***Via email**

OFFICE	NRR/JLD/JOMB/PM	NRR/JLD/LA	NRR/JLD/JERB/BC*
NAME	JBoska	SLent	SBailey
DATE	09/25/15	09/23/15	10/16/15
OFFICE	NRR/JLD/JCBB/BC(A)*	NRR/JLD/JHMB/BC*	NRR/JLD/JOMB/BC(A)
NAME	BTitus	MShams	MHalter
DATE	10/16/15	10/04/15	11/9/2015

OFFICIAL RECORD COPY