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

September 20, 2017

Mr. John Dent, Jr.
Vice President-Nuclear and CNO
Nebraska Public Power District
Cooper Nuclear Station
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SUBJECT: COOPER NUCLEAR STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 (CAC NOS. MF0971 AND MF0972)

Dear Mr. Dent:

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

By letter dated February 28, 2013 (ADAMS Accession No. ML13070A007), Nebraska Public Power District (NPPD, the licensee) submitted its OIP for Cooper Nuclear Station (CNS) in response to Order EA-12-049. At six month intervals following the submittal of its OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 11, 2014 (ADAMS Accession No. ML14007A650), and August 29, 2016 (ADAMS Accession No. ML16217A475), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated January 4, 2017 (ADAMS Accession No. ML17017A166), NPPD submitted a compliance letter and Final Integrated Plan (FIP) in response to Order EA-12-049. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-049.

By letter dated February 28, 2013 (ADAMS Accession No. ML13070A007), the licensee submitted its OIP for CNS in response to Order EA-12-051. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated December 4, 2013 (ADAMS Accession No. ML13323A105), and August 29, 2016 (ADAMS Accession No. ML16217A475), 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 20, 2016 (ADAMS Accession No. ML17234A314), CNS submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051.

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

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

Sincerely,

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

Docket No.: 50-298

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12054A736), the NRC issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events". 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 (ADAMS Accession No. ML12054A679), the NRC also issued Order EA-12-051, "Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation". This order directed licensees to install reliable SFP level instrumentation with a primary channel and a backup channel, and with independent power supplies that are independent of the plant alternating current (ac) and direct current (dc) power distribution systems. Order EA-12-051 applies to all power reactor licensees and all holders of construction permits for power reactors.

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC regulations and processes and determining if the agency should make additional improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the

NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 (ADAMS Accession No. ML11186A950). Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012, the NRC staff provided SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami" (ADAMS Accession No. ML12039A103), to the Commission. This paper included a proposal to order licensees to implement enhanced BDBEE mitigation strategies. As directed by the Commission in staff requirements memorandum (SRM)-SECY-12-0025 (ADAMS Accession No. ML120690347), the NRC staff issued Orders EA-12-049 and EA-12-051.

2.1 Order EA-12-049

Order EA-12-049, Attachment 2 (ADAMS Accession No. ML12054A736), requires that operating power reactor licensees and construction permit holders use a three-phase approach for mitigating BDBEEs. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and SFP cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely. Specific requirements of the order are listed below:

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

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

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

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

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

On December 10, 2015 (ADAMS Accession No. ML16005A625), following submittals and discussions in public meetings with NRC staff, the Nuclear Energy Institute (NEI) submitted document NEI 12-06, Revision 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide", to the NRC to provide revised specifications for an industry-developed methodology for the development, implementation, and maintenance of guidance and strategies in response to the Mitigation Strategies order. The NRC staff reviewed NEI 12-06, Revision 2, and on January 22, 2016 (ADAMS Accession No. ML15357A163), issued Japan Lessons-Learned Division (JLD) Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 1, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events", endorsing NEI 12-06, Revision 2, with exceptions, additions, and clarifications, as an acceptable means of meeting the requirements of Order EA-12-049, and published a notice of its availability in the *Federal Register* (81 FR 10283).

2.2 Order EA-12-051

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

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

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

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

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

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

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

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

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

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

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

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

On August 24, 2012 (ADAMS Accession No. ML12240A307), 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 to the NRC to provide specifications for an industry-developed methodology for compliance with Order EA-12-051. On August 29, 2012 (ADAMS Accession No. ML12221A339), the NRC staff issued its final version of JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation", endorsing NEI 12-02, Revision 1, as an acceptable means of meeting the requirements of Order EA-12-051 with certain clarifications and exceptions, and published a notice of its availability in the *Federal Register* (77 FR 55232).

3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 (ADAMS Accession No. ML13070A009), Nebraska Public Power District (NPPD, the licensee) submitted its Overall Integrated Plan (OIP) for Cooper Nuclear Station (CNS, Cooper) in response to Order EA-12-049. By letters dated August 27, 2013 (ADAMS Accession No. ML13247A283), February 26, 2014 (ADAMS Accession No. ML14064A201), August 26, 2014 (ADAMS Accession No. ML14246A188), February 23, 2015 (ADAMS Accession No. ML15062A040), August 27, 2015 (ADAMS Accession No. ML15251A239), February 16, 2016 (ADAMS Accession No. ML16054A799), and August 26, 2016 (ADAMS Accession No. ML16245A289), the licensee submitted six-month updates to its OIP. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 11, 2014 (ADAMS Accession No. ML14007A650), and August 29, 2016 (ADAMS Accession No. ML16217A475), the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated January 04, 2017 (ADAMS Accession No. ML17017A166), the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP).

3.1 Overall Mitigation Strategy

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

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

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

Cooper is a General Electric (GE) boiling-water reactor (BWR) Model 4 with a Mark I containment. The licensee's three-phase approach to mitigate a postulated ELAP event, as described in its FIP, is summarized below.

At the onset of an ELAP, the reactor is assumed to trip from full power. The main condenser is unavailable due to the loss of circulating water. Decay heat is removed when the safety relief valves (SRVs) open on high pressure and dump steam from the reactor pressure vessel (RPV) to the suppression pool located in the containment. After the initial transient, makeup to the RPV is provided by the reactor core isolation cooling (RCIC) turbine-driven pump. The RCIC pump takes suction from the emergency condensate storage tanks (ECSTs) with a backup supply line from the suppression pool. Within 30 minutes, the operators take manual control of the SRVs to perform a controlled cooldown and depressurization of the reactor. The cooldown of the primary system is stopped when reactor pressure reaches a control band of 150 per square inch gauge (psig) to 300 psig to ensure sufficient steam pressure to operate the RCIC pump. When the suppression pool heats up to a predetermined setpoint, the containment vent is opened to mitigate the temperature rise and allow the RCIC system to continue to function. The RCIC injection source will be maintained for as long as possible, since the suction sources of ECST and/or suppression pool are maintained using relatively clean water.

The primary strategy for RPV makeup during Phase 2 is the continued use of the RCIC pump taking suction from the ECST. When the ECST water becomes depleted, the ECST is refilled first from the hotwell, if available, and then, from an on-site well. Installed pumps will either transfer condensate from the hotwell or the on-site well to the ECST through a combination of hoses and installed piping. As an alternate strategy, a FLEX pump is deployed which can provide RPV injection of well water through one of two connections. The primary connection is via the normal residual heat removal service water (RHRSW) crosstie to the Division 1 residual heat removal (RHR) injection flow path. An alternate connection uses the same portable FLEX pump but connects through the Division 2 RHR torus spray line to the Division 2 RHR injection flow path.

The reactor has a Mark I containment which is inerted with nitrogen at power. The licensee performed a containment evaluation and determined that containment integrity is maintained by normal design features of the containment, such as the containment isolation valves and the hardened containment venting system (HCVS). Opening the HCVS at a containment pressure of approximately 15 psig will allow containment temperature and pressure to stay within acceptable levels.

The reactor has a SFP in its reactor building. SFP cooling is provided by adding water to makeup for boil-off. There are no Phase 1 actions required for SFP makeup; the long time to boil enables deployment of Phase 2 equipment for this function. The normal SFP water level at the event initiation provides for at least 21 feet (ft.) 6 inches. (in.) of water inventory above the top of the stored spent fuel. The most limiting time to fuel uncovery is 44.82 hours resulting from a full-core offload, 5 days after shutdown. For other Modes, 155.82 hours (6.49 days) are available to fuel uncovery.

To makeup to the SFP during Phase 2, the licensee has a primary and alternate strategy, in which operators begin staging equipment at 24 hours into the event. The primary strategy uses the portable FLEX pump to tie into a fuel pool cooling (FPC) system chemical decontamination connection. Valves are aligned to supply makeup through the FPC System to the SFP through the normal fill location. The alternate strategy employs a portable FLEX pump and hoses.

The operators will perform dc bus load stripping within the initial 4 hours following event initiation to ensure safety-related battery life is extended up to 9 hours. Following dc load stripping and prior to battery depletion, one 175-kilowatt (kW), 480 volt alternating current (Vac) will be deployed from a FLEX storage building (FSB). This portable generator will be used to repower one division of essential battery chargers within 5 hours of ELAP initiation.

For Phase 3, a National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) will provide high capacity pumps and large combustion turbine-driven generators. This equipment can be used to supplement or replace Phase 2 portable equipment or to restore one RHR cooling train to cool the core for long-term Phase 3 actions. There are two NSRCs in the United States.

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

3.2 Reactor Core Cooling Strategies

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

As reviewed in this section, the licensee's core cooling analysis presumes that, per endorsed guidance from NEI 12-06, the unit would have been operating at full power prior to the event. Therefore, the suppression pool may be credited as the heat sink for core cooling. Maintenance of sufficient RPV inventory, despite steam release from the SRVs and ongoing system leakage expected under ELAP conditions, is accomplished through a combination of installed systems and FLEX equipment. The specific means used by the licensee to accomplish adequate core cooling is discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions during shutdown or refueling conditions is reviewed separately in Section 3.11 of this evaluation.

3.2.1 Core Cooling Strategy and RPV Makeup

3.2.1.1 Phase 1

In its FIP, the licensee stated that the injection of cooling water into the RPV will be accomplished through the RCIC system. The RCIC system suction is initially lined up to the ECSTs and will pump water into the core automatically. The ECSTs are located inside Class 1 structures and are robust for all external hazards, and thus are credited to be available in all FLEX scenarios. If the ECSTs water is depleted, RCIC suction will automatically transfer to the suppression pool based on a level signal from the ECSTs. The suppression pool is also fully protected from all external hazards.

The RCIC pump is powered by a turbine using steam from the RPV and is robust for all external hazards. Both the RCIC and HPCI pumps are designed to automatically start following the ELAP event. Following the initial restoration of RPV water level, operators will secure the HPCI system. In the event that RCIC does not automatically start, procedural guidance directs the operators to manually start the pump. The RCIC discharges into the reactor feedwater injection header of the RPV. The RCIC system valves are powered by the 125 volt direct current (Vdc) bus and are used to control the cooling flow to the RPV, balancing it with the outflow of steam through the SRVs in order to maintain the RPV level within its desired control band.

Pressure control of the RPV is accomplished using the SRVs, which are powered by the station batteries. The SRVs normal pneumatic supply is lost at event initiation due to the loss of power. Until the reactor building air header is repressurized by a portable diesel-driven air compressor during Phase 2, the SRVs will be supplied with pneumatic pressure maintained by accumulators associated with the SRVs and automatic depressurization system (ADS) valves. During initial Phase 1, the SRVs will be used to maintain RPV pressure between approximately 800 and 1000 psig in accordance with CNS procedures. Then after approximately 30 minutes, the operators take manual control of the SRVs to perform a controlled cooldown and depressurization of the reactor. The cooldown of the primary system is stopped when reactor pressure reaches a control band of 150 psig to 300 psig to ensure sufficient steam pressure to operate the RCIC pump.

After the start of the ELAP, SRV blowdown and RCIC pump exhaust will cause the torus temperature to rise to a point where containment pressure will rise. The licensee will use the hardened containment venting system (HCVS) to maintain containment pressure between 5 and 15 psig. This pressure band will allow continued operation of RCIC. This venting will maintain the torus water temperature less than 250°F until other means of containment cooling become available. The licensee expects to open the vent at approximately 8 hours after the initiation of the ELAP event. The vent system is provided with power from two uninterruptable power supply (UPS) systems. One UPS system is dedicated to the operation of the motor operated valve, which will need to be opened only once. The other UPS provides power for controls and indications with the associated HCVS system. The pneumatically operated valves in the vent system have accumulators to provide air for the first 24 hours, and can be provided with nitrogen for at least a week of operation.

The SRVs are safety-related equipment, which are located inside the plant drywell and protected from all external hazards. They have backup air accumulators, which have sufficient capacity to maintain adequate pressure for SRV operation throughout an ELAP event. The SRVs are powered from the Class 1E 125 Vdc distribution system. The licensee has stated that the ADS is qualified at a maximum temperature of 340°F and the licensee's Modular Accident Analysis Program (MAAP) analysis indicates that the environmental conditions reached during the ELAP event (a steady state temperature of approximately 250°F) in the drywell will not challenge the continued operation of the SRVs.

The CNS batteries and Class 1E 125 Vdc distribution system provide power to the RCIC system, SRVs and required instrumentation. A load shed of the direct current (dc) system is completed approximately 4 hours after the initiation of the ELAP event to extend the battery capacity to power the Phase I systems and instruments. Installed batteries can maintain necessary voltage for at least 9 hours. Prior to battery depletion, a FLEX diesel generator (DG) is deployed and used to recharge either the Division 1 or Division 2 batteries.

3.2.1.2 Phase 2

In its FIP, the licensee stated that the preferred method for maintaining core cooling will be by utilizing the RCIC system. The suction for the RCIC system will remain aligned to the ECSTs. As the ECSTs are depleted, they will be refilled from the condenser hotwell, if available, and then from an on-site well. The well is robust and sized at 262 gallons per minute (gpm) for providing the makeup required. Power for the hotwell pump and the well will be provided by a portable FLEX DG rated for 60 kW. Hoses will be connected from the discharge of either the hotwell pump or the well to a connection point on the ECSTs.

A FLEX pump also can be used to provide injection to the RPV. The FLEX pump is rated for 925 gallons per minute (gpm) at 378 ft. of head. The FLEX pump will take suction from the discharge of the well pump and discharge to one of two connection points. The primary connection is a FLEX connection located on the residual heat removal service water system and will discharge to the RPV via the Division 1 RHR injection flow path. The alternate connection point is the existing credited B.5.b connection (i.e., installed under 10 CFR 50.54(hh)(2) implementation) on the RHR torus spray line and will discharge to the RPV via the Division 2 RHR injection flow path.

In order to maintain the continued use of the SRVs, RCIC and required instrumentation, a 175 kW FLEX generator will be connected to FLEX connections to supply power to one division of the plant battery chargers within 5 hours. The plant batteries are sized to power the dc systems for approximately 9 hours without recharging, providing for four hours of margin.

3.2.1.3 Phase 3

The Phase 3 strategy includes the use of equipment from the NSRC. The plant plans to continue the use of Phase 2 equipment or replace as necessary. In addition, the licensee plans to utilize one RHR loop in the shutdown cooling mode of operation. Two 4160 volt alternating current (Vac) generators provided by the NSRC will be utilized to restore power to either the 4160 Vac Bus F or 4160 Vac Bus G, providing power to a RHR pump. The UHS function will be restored by a large pump provided by the NSRC which will take suction from the Missouri River and supply water to the RHR service water connection point. Following the restoration of 4160 Vac equipment, the reactor equipment cooling system will be restored to provide cooling to the associated RHR pump bearing and seal oil cooler as well as the RHR pump room cooler.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

The design-basis flood elevation at CNS is 903 ft. mean sea level (MSL). In its FIP, the licensee stated that the floor elevation of the FSB and Class I structures are at 903.5 ft. MSL. The licensee stated that flooding of the station is extremely unlikely because of the flood control measures located upstream on the Missouri River and the high site grade. In addition, the local site drainage can remove accumulated water from probable maximum precipitation (PMP). Effects of on-site flooding are discussed in Section 3.5.2 of this safety evaluation. The licensee's core cooling and makeup strategy implementation remain the same for a flooding event; therefore, no variations to the core strategy are necessary at this time.

3.2.3 Staff Evaluations

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

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

3.2.3.1.1 Plant SSCs

Phase 1

In the FIP, Section 2.3.1 states that the primary strategy for core cooling is to supply high quality water via the RCIC system. The RCIC system consists of a steam-driven turbine pump that takes suction from the ECSTs or uses the suppression pool as a backup water supply. The system is capable of remote manual initiation from the main control room; however, it is designed to operate automatically, in which case only dc power from the station's 125-volt batteries is needed to operate valves and controls. As described in FIP Section 2.3.4.1, the RCIC system, suppression pool, and ECSTs are located in the reactor building and control building, which are Class I structures designed to be robust with respect to a Safe Shutdown Earthquake (SSE), wind loading, tornado loading, and tornado generated missiles. In addition, the RCIC system and its instrumentation and controls are designed to seismic Class I specifications; therefore, the NRC staff concludes that the RCIC system including the steam driven turbine-pump, ECSTs, and suppression pool 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.

During decay heat removal, the primary strategy for reactor pressure control is by operation of the SRVs. These valves require dc control power from the station's batteries to operate. In addition, they require pneumatic pressure for operation. In the FIP, Sections 2.3.1 and 2.3.4.2 state that the normal pneumatic supply is lost due to the loss of ac power. Following the loss of the normal supply, pneumatic accumulators supply pressure. As stated in FIP Section 2.3.4.2, the accumulators have a combined capacity sufficient to allow the SRVs to operate for at least 24 hours, or until Phase 2, when a pneumatic supply is restored using FLEX equipment. The SRVs and controls are located in the reactor building and control building described above as Class I structures. The NRC staff concludes that the SRVs 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.

Phase 2

The licensee's Phase 2 strategy continues to use the suppression pool as the heat sink for SRV discharges and RCIC turbine steam exhaust. The RCIC will continue to be used in automatic mode with suction from the ECSTs until shutdown cooling is placed in service in Phase 3. The ECSTs will be refilled from the hotwell first, if available, and then from an on-site well. In the FIP, Section 2.3.2 states that a FLEX pump will be deployed which can inject well water into the RPV through primary or alternate connection points. The licensee plans to rely on the FLEX connection points and water sources discussed below in safety evaluation Sections 3.7 and 3.10, respectively. In addition, the SRVs will continue to be used for pressure control. As described in FIP Sections 2.3.2 and 2.3.4.2, a FLEX portable diesel-driven air compressor will be connected to the reactor building reliable air header to provide a pneumatic source for the SRVs.

Phase 3

The plant plans to continue the use of Phase 2 equipment or replace as necessary. In addition, the licensee plans to place one loop of the RHR and service water into shutdown cooling (SDC) mode. As described in FIP Section 2.3.3, this is accomplished by powering a Division 1 (or alternatively Division 2) RHR pump using two portable 4160 Vac combustion turbine generators (CTGs) from the NSRC and restoring the UHS function by supplying river water to the RHR heat exchanger from an NSRC portable pump. In the updated final safety analysis report (UFSAR) Section 2.1.2.3, indicates that the RHR and service water pumps and heat exchangers are seismic Class I components and are located in the RHR pump room, which is a Class I structure. The RHR pump room is designed to be robust with respect to an SSE, wind loading, tornado loading, and tornado generated missiles. Therefore, the NRC staff concludes that the RHR pumps and heat exchangers 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.

3.2.3.1.2 Plant Instrumentation

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

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

- RPV level (fuel zone)
- RPV level (wide, narrow, and nozzle range)
- RPV pressure
- Containment/torus level(wide range)
- Torus level
- Torus temperature
- Torus pressure
- Drywell pressure
- Drywell temperature
- ECST level

These instruments are monitored from the control room, in the control building and local racks.

The instrumentation identified by the licensee to support its core cooling strategy is consistent with the recommendation specified in the endorsed guidance of NEI 12-06.

Per the FIP, instrumentation is normally powered by station batteries. Load shedding of non essential equipment extends the battery power up to 9 hours without charging. One division of the 125 and 250 Vdc batteries are charged within 5 hours of the ELAP event by a FLEX 175 kW DG, which provides power to the battery chargers in Phases 2 and 3. Additional backup generators will be available from the NSRC during Phase 3. Therefore, based upon the information provided by the licensee, the NRC staff understands that the locations of the instrument indications would be accessible continuously throughout the ELAP event.

The FIP stated that guidelines to control critical equipment (RCIC and SRVs) without control power are provided in Procedure 5.3 "Alternate Strategy." In accordance with NEI 12-06 Section 5.3.3.1, CNS procedure 5.10 FLEX.28, "Vital Instrumentation FLEX Operations," provides guidance on powering portable instruments from alternate power supplies to allow critical readings to be obtained when normal instrument ac power is not available. Some critical parameters in the procedure include reactor level nozzle range; RPV wide range and fuel zone level, and containment, drywell, and torus pressure and level. The SFP level instruments are discussed in Section 4 below.

3.2.3.2 <u>Thermal-Hydraulic Analyses</u>

The licensee based its mitigating strategy for reactor core cooling in part on thermal-hydraulic analysis performed using Version 4 of the MAAP (MAAP4). Because the thermal-hydraulic analysis for the reactor core and containment during an ELAP event are closely intertwined, as is typical of BWRs, the licensee has addressed both in a single, coupled calculation. This dependency notwithstanding, the NRC staff's discussion in this section of the safety evaluation solely focuses on the licensee's analysis of reactor core cooling. The review of the licensee's analysis of containment thermal-hydraulic behavior is provided in Section 3.4.4.2 of this evaluation.

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

To provide analytical justification for their mitigating strategies in response to Order EA-12-049, a number of licensees for BWRs and pressurized-water reactors (PWRs) completed analysis of the ELAP event using MAAP4. Although MAAP4 and predecessor code versions have been used by industry for a range of applications, such as the analysis of severe accident scenarios and probabilistic risk analysis (PRA) evaluations, the NRC staff had not previously examined the code's technical adequacy for performing best-estimate simulations of the ELAP event. In particular, due to the breadth and complexity of the physical phenomena within the code's calculation domain, as well as its intended capability for rapidly simulating a variety of accident scenarios to support PRA evaluations, the NRC staff observed that the MAAP code makes use of a number of simplified correlations and approximations that should be evaluated for their applicability to the ELAP event. Therefore, in support of the reviews of licensees' strategies for ELAP mitigation, the NRC staff audited the capability of the MAAP4 code for performing thermal-hydraulic analysis of the ELAP event for both BWRs and PWRs. The NRC staff's audit review involved a limited review of key code models, as well as confirmatory analysis with the TRACE code to obtain an independent assessment of the predictions of the MAAP4 code.

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

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

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

During the audit process, the NRC staff verified that the licensee's MAAP4 calculation, along with an associated addendum, addressed the limitations from the NRC staff's endorsement letter. The licensee utilized the generic roadmap and response template that had been developed by EPRI to support consistency in individual licensee's responses to the limitations from the endorsement letter. In particular, based upon review of the MAAP4 calculation documentation, the staff concluded that appropriate inputs and modeling options had been selected for the code parameters expected to have dominant influence for the ELAP event. The NRC staff further observed that the limitations imposed in the endorsement letter, particularly those concerning the RPV collapsed liquid level being maintained above the reactor core and the primary system cooldown rate being maintained within technical specification limits, were satisfied. Specifically, the licensee's analysis calculated that CNS would maintain the collapsed liquid level in the reactor vessel above the top of the active fuel region throughout the analyzed ELAP event. The licensee calculated that the minimum RPV water level above the top of active fuel is approximately 8 ft. and occurs during the initial RPV depressurization. By maintaining the reactor core fully covered with water, adequate core cooling is assured for this event. Additionally, CNS' fulfillment of the endorsement letter condition regarding the primary system cooldown rate signifies that thermally induced volumetric contraction and other changes in primary system thermal-hydraulic conditions should proceed relatively slowly with time, which supports the NRC staff's confidence in the predictions of the MAAP4 code. Furthermore, the licensee's ability to maintain the entire reactor core submerged throughout the ELAP event is consistent with the staff's expectation that the licensee's flow capacity for primary makeup (i.e., installed RCIC pump and subsequently, FLEX pumps) should be sufficient to support adequate heat removal from the reactor core during the analyzed ELAP event, including potential losses due to expected primary leakage.

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

3.2.3.3 Recirculation Pump Seals

An ELAP event would result in the interruption of cooling to the recirculation pump seals, potentially resulting in increased leakage due to the distortion or failure of the seals, elastomeric O-rings, or other components. Sufficient primary make-up must be provided to offset recirculation pump seal leakage and other expected sources of primary leakage, in addition to removing decay heat from the reactor core.

The licensee's calculations for CNS assumed a total leakage rate at normal RPV operating pressure of 66 gpm. This leakage rate includes 18 gpm per recirculation pump seal, which is in accordance with work performed to support NRC Generic Letter 91-07. In addition, the licensee's calculation assumed a primary system leakage rate equal to the technical specification LCO 3.4.4 limit of 30 gpm. Thus, between the two recirculation pumps and the additional primary system leakage, the total primary leakage rate assumed for CNS during the ELAP event was 66 gpm at normal operating reactor pressure. This leakage rate was used in CNS MAAP4 analysis.

Considering the above factors, the NRC staff concludes that the leakage rate of 66 gpm is reasonable. Gross seal failures are not anticipated to occur during the postulated ELAP event. As is typical of the majority of BWRs, CNS has an installed steam-driven pump (i.e., RCIC) capable of injecting into the primary system at a flow rate well in excess of the primary system leakage rate expected during an ELAP event, and the other pumps used for core cooling in its FLEX strategy have a similar functional capability. As discussed previously, at the limiting pressure, the FLEX pump is able to inject at a rate which maintains adequate margin.

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

3.2.3.4 Shutdown Margin Analyses

As described in the CNS UFSAR, the control rods provide adequate shutdown margin under all anticipated plant conditions, with the assumption that the highest-worth control rod remains fully withdrawn. CNS technical specification Section 1.1, Definitions, further clarifies that shutdown margin is to be calculated for a cold, xenon-free condition to ensure that the most reactive core conditions are bounded.

Based on the NRC staff's audit review, the licensee's ELAP mitigating strategy maintains the reactor within the envelope of conditions analyzed by the licensee's existing shutdown margin calculation. Furthermore, the existing calculation retains conservatism because the guidance in NEI 12-06 permits analyses of the beyond-design-basis ELAP event to assume that all control rods fully insert into the reactor core.

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

3.2.3.5 FLEX Pumps and Water Supplies

In the FIP, Section 3.2.3 states that RCIC is maintained as the primary strategy for core cooling until Phase 3 when shutdown cooling is implemented. The licensee has an alternate strategy as defense-in-depth that utilizes one portable diesel-driven pump during Phase 2. In the FIP, Sections 2.3.9 and 2.18.5 state the performance criteria (e.g., flow rate and head) for this pump. See safety evaluation Section 3.10 for a detailed discussion of the availability and robustness of each water source.

If RCIC is not available, the FLEX pump is used to inject well water into the RPV from primary or alternate connection points. A FLEX pump is stored in each of the two FSBs. As described in

FIP Section 2.3.9, this pump takes suction from the installed North well pump which is located in an enclosed structure capable of withstanding tornado missiles and the design-basis earthquake. The FLEX pump is rated at 925 gpm at 378 ft. head. To verify this pump capacity is within requirements, the licensee performed calculation NEDC 15-002, "Portable Equipment Calculations in support of CNS FLEX Strategy," Revision 1. Calculation NEDC 15-002, Revision 1 is a hydraulic analysis, and it determined for various RPV pressures, the volumetric flow rate and head needed to remove decay heat following a BDBEE.

The staff confirmed that the FLEX pump capacity was within the required capacity from the hydraulic analysis. During the onsite audit, the staff conducted a walk down of the hose deployment routes for the FLEX pump to confirm the evaluations of the pump staging locations, hose distance runs, and connection points as described in the hydraulic analysis and FIP.

Based on the staff's review of the FLEX pumping capabilities at CNS, the licensee has demonstrated that its FLEX pump should perform as intended to support core cooling and RCS makeup during an ELAP caused by an external event, consistent with NEI 12-06, Section 11.2.

3.2.3.6 Electrical Analyses

The licensee's electrical strategies provide power to the equipment and instrumentation used to mitigate the ELAP and LUHS. The electrical strategies described in the FIP are practically identical for maintaining or restoring core cooling, containment, and SFP cooling, except as noted in Sections 3.3.4.4 and 3.4.4.4 of this safety evaluation.

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

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

During the first phase of the ELAP event, CNS would rely on the Class 1E station batteries to provide power to key instrumentation for monitoring parameters and power to controls for SSCs used to maintain the key safety functions (reactor core cooling, RCS inventory control, and containment integrity). The CNS Class 1E station batteries and associated dc distribution systems are located in a safety-related structure designed to meet all applicable design-basis external hazards. The licensee's procedure 5.3SBO, "Station Blackout," would direct operators to conserve dc power during the event by stripping non-essential loads. Operators will strip or shed unnecessary loads to extend battery life until backup power (Phase 2) is available. The plant operators would commence load shedding within 1 hour and complete load shedding within 4 hours from the onset of an ELAP event.

Cooper has two Class 1E, 125 Volt (V) dc batteries (Division 1 and 2) and two Class 1E, 250 Vdc batteries (Division 1 and 2). The Class 1E batteries are located in two rooms, each containing a 125 Vdc and a 250 Vdc battery. The Class 1E station batteries were manufactured by C&D Technologies. The Class 1E station batteries are model LCR-25 rated at 1715 ampere-hours at an 8-hour discharge rate to 1.81 V per cell. The NRC staff reviewed the licensee's dc sizing calculation (NEDC 87-131A, "250 VDC Division 1 Load and Voltage Study," Revision 13C1, NEDC 87-131B, "250 VDC Division 2 Load and Voltage Study," Revision 12C2, NEDC 87-131C, "125 VDC Division 1 Load and Voltage Study," Revision 15C1, and NEDC 87-131D, "125 VDC Division 2 Load and Voltage Study," Revision 13C13) which verified the capability of the dc system to supply power to the required loads (i.e., the dc buses, and three ventilation fans) during the first phase of the CNS FLEX mitigation strategy plan for an ELAP event. The licensee's evaluation identified the required loads and their associated ratings (ampere (A) and minimum required voltage) and the non-essential loads that would be shed within 4 hours to ensure battery operation until power is expected to be restored to one division of the 125/250 Vdc battery chargers (approximately 5 hours). Based on its review, the NRC staff confirmed that the 125 Vdc Division 1 battery could be extended up to approximately 9 hours, the 125 Vdc Division 2 battery could be extended up to 10.5 hours, the 250 Vdc Division 1 battery could be extended up to 9.5 hours, and the 250 Vdc Division 2 battery could be extended up to 12 hours by shedding non-essential loads.

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

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

The licensee's Phase 2 strategy includes repowering the 125/250 Vdc battery chargers C and either division of dc buses approximately 5 hours after initiation of an ELAP event. The licensee's strategy relies on one portable 175 kW 480 Vac FLEX DG (N). The licensee has two portable 175 kW 480 Vac FLEX DGs to provide N+1 for this function.

The licensee's Phase 2 strategy also includes deploying a 60 kW 480 Vac FLEX DG in support of its core cooling and HCVS. The 60 kW FLEX DG would supply a hotwell transfer pump, makeup well, and an HCVS battery charger (to maintain the ability to open and close valves). The licensee has two portable 60 kW, 480 Vac FLEX DGs to provide N+1 for this function.

The NRC staff reviewed the licensee's calculation CED 6037041, "FLEX Electrical Connections," Revision 0, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the EDGs. Based on the NRC staff's review, the required loads for the Phase 2 175 kW FLEX DG is approximately 151.1 kW. The required loads for the Phase 2, 60 kW FLEX DG is approximately 43.4 kW. Therefore, one 175 kW and one 60 kW FLEX DG is adequate to support the electrical loads required for the licensee's Phase 2 strategy.

For Phase 3, the licensee plans to utilize additional assistance provided from offsite equipment/resources. The offsite resources that will be provided by an NSRC include two 1-megawatt (MW), 4160 Vac combustion turbine generators (CTGs); one 1100 kW, 480 Vac CTG; and distribution panels (including cables and connectors). Each portable 4160 Vac CTG is capable of supplying approximately 1 MW, but two CTGs could be operated in parallel to provide a total of approximately 2 MW per unit. The NRC staff reviewed the licensee's calculation CED 6037041, FSG5.10FLEX.08, "4160V "G" Bus Tie-In with Off-Site Generator," Revision 1, and procedures FSG 5.10FLEX.07, "4160V "F" Bus Tie-In with Off-Site Generator," Revision 1, FSG 5.10FLEX.14, "RHR Div 1 Shutdown Cooling FLEX Operations," Revision 1, and FSG 5.10FLEX.15, "RHR Div 2 Shutdown Cooling FLEX Operations," Revision 0.

Based on calculations and documents, the continuous loads on the selected division would total approximately 1096.5 kW. Therefore, two 4160 Vac CTGs have sufficient capacity to supply all of the loads on 4160 Vac bus F or G necessary to support the licensee's Phase 3 FLEX strategies, which includes an RHR pump and its support equipment (i.e., MOVs, reactor equipment cooling (REC) system, room coolers, etc.). While the licensee does not plan to use the NSRC supplied 480 Vac CTG, if could be used as a replacement for the Phase 2 175 kW or 60 kW FLEX DG, if necessary, since it is of larger capacity than the onsite Phase 2 FLEX DGs.

Based on its review of this calculation and the licensee's FIP, the NRC staff concludes that the NSRC supplied 4160 Vac and 480 Vac CTGs have adequate capacity to supply the Phase 3 loads.

3.2.4 Conclusions

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

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Table 3-1 and Appendix C summarize an approach consisting of two 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; and 2) makeup via connection to SFP cooling piping or other alternate location capable of exceeding the boil-off rate for the design-basis heat load. However, in JLD-ISG-2012-01, Revision 1, the NRC staff did not fully accept this approach, and added another requirement to either have the capability to provide spray flow to the SFP, or complete an SFP integrity evaluation which demonstrates that a seismic event would have a very low probability of inducing a crack in the SFP or its piping systems so that spray would not be needed to cool the spent fuel. The evaluation must use the reevaluated seismic hazard described in Section 3.5.1 below if it is higher than the site's current SSE. During the event, the licensee selects the SFP makeup method to use based on plant conditions. This approach also requires a strategy to mitigate the effects of steam from the SFP, such as venting.

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

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

The ELAP causes a loss of cooling in the SFP. As a result, the pool water will heat up and eventually boil off. The licensee's response is to provide makeup water. The timing of operator actions and the required makeup rates depend on the decay heat level of the fuel assemblies in the SFP. The sections below address the response during operating, pre-fuel transfer or post-fuel transfer operations. The effects of an ELAP with full core offload to the SFP is addressed in Section 3.11 of this safety evaluation. The licensee performed the SFP integrity analysis to demonstrate that SFP spray flow is not needed. The staff reviewed the licensee's evaluation and found it was in accordance with NRC approved guidance as stated in letter dated March 3, 2017 (ADAMS Accession No. ML16313A282).

3.3.1 Phase 1

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

3.3.2 Phase 2

In the FIP, Section 2.4.2 states that operators will deploy a portable FLEX pump to supply water to the SFP during Phase 2 at rates greater than the SFP boil off rate of 70 gpm. Phase 2 equipment will be staged within 24 hours after the ELAP event. The licensee's primary strategy will use the FLEX pump to discharge to a fuel pool cooling system chemical decontamination connection. Valves will be aligned to supply the makeup water to the SFP. Alternatively, the portable FLEX pump will be used, and hoses will directly provide make up to the SFP.

3.3.3 Phase 3

In the FIP, Section 2.4.3 states that the Phase 3 SFP cooling strategy is a continuation of the Phase 2 strategy.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

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

In the FIP, Section 2.17 (i.e., the licensee's sequence of events timeline) shows that operators will stage a portable pump and begin makeup to the SFP at approximately 24 hours into the BDBEE. In the FIP, Section 2.4.1 states that the licensee's Phase 1 SFP cooling strategy does not require any operator actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of water in the SFP. In the FIP, Section 2.11.1.2 states that operators are directed to open the reactor building roof hatch to prevent condensation and steam buildup. Additionally, operators are directed to open doors at the ground level, and doors on the 903 Mezzanine, to establish a ventilation path.

The licensee's Phase 2 and 3 SFP cooling strategies credit the use of a hardened well, a 20,000 gallon bladder as a surge volume, and a portable FLEX pump (see FIP Section 2.4.4.1.). The FLEX pump supplies water to the SFP via the fuel pool cooling system or discharges directly to the SFP through a 5 inch hose (see Section 3.3.4.3 of this safety evaluation for additional discussion of FLEX pumps). The SFP level and temperature are monitored using installed instrumentation. In addition, ventilation for the SFP remains aligned as previously established through an opened roof hatch, airlock, and doors. The evaluation of the robustness and availability of FLEX connection points for the FLEX pump is discussed in Section 3.7.3.1 of this safety evaluation. Furthermore, the evaluation of the robustness and availability of the UHS for an ELAP event is discussed in Section 3.10.3 of this safety evaluation.

3.3.4.1.2 Plant Instrumentation

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

3.3.4.2 Thermal-Hydraulic Analyses

In the FIP, Section 2.4.6 describes two scenarios: (1) a power operation heat load scenario, 30 days after shutdown in which the pool contains 220 recently offloaded bundles; and (2) a refueling heat load scenario for a full core offload, 5 days after shutdown. The SFP will boil in approximately 13.29 hours after initiation of the ELAP event for normal power operation (or non-outage) conditions and 3.82 hours for a full core offload condition. For the normal power operation SFP heat load, fuel uncovery occurs 155.8 hours into the event and 44.8 hours into the event for the full core offload condition.

As described in Section 6.1.2 of NEDC 15-002, Revision 1, "Portable Equipment Calculations in support of CNS FLEX Strategy," the licensee determined that a SFP makeup rate of 70 gpm is adequate to maintain SFP level. Consistent with the guidance in NEI 12-06, Section 3.2.1.6, the staff concludes that the licensee has considered the maximum design-basis SFP heat load.

3.3.4.3 FLEX Pumps and Water Supplies

As described in FIP Sections 2.4.2 and 2.4.7, the SFP cooling strategy relies on the combination of the FLEX pump and North well pump to provide SFP makeup during Phase 2. The North well pump provides the suction source for the FLEX pump, either directly or with a bladder, to provide makeup to the SFP. The discharge of the FLEX pump is directed to the SFP either through a connection on the fuel pool cooling system or directly through a hose. In the FIP, Sections 2.3.9 and 2.4.7 describe the hydraulic performance criteria (e.g., flow rate and head) for the FLEX pump and the North well pump. As stated in NEDC 15-002, "Portable Equipment Calculations in support of CNS FLEX Strategy," Revision 1, a pump flow rate of 255 gpm with a discharge head of 207.7 ft is required to fill the SFP to the initial water level (i.e., 36 ft) within 5 hours, and thereafter, requires 70 gpm at 206.5 ft. of head to maintain the initial pool level. The FLEX pump (925 gpm at 378 ft. head) in combination with the North well pump (262 gpm at 226 ft. head) can provide the required SFP flow rate that meets and exceeds the maximum SFP makeup requirements. The NRC staff notes that the performance criteria for the RPV pump supplied by the NSRC for Phase 3 shows that it would fulfill the mission of the onsite portable FLEX pump if it were to fail. Furthermore, the NRC staff noted that the analysis in NEDC 15-002, Revision 1 is consistent with NEI 12-06 Section 11.2. Therefore, the NRC staff concludes that the FLEX equipment is capable of supporting the SFP cooling strategy, and it is expected to be available during an ELAP event.

3.3.4.4 Electrical Analyses

The licensee's mitigating strategies for the SFP do not rely on electrical power except for power to SFP level instrumentation. The licensee's electrical SFP cooling strategy for all phases is to monitor SFP level using installed instrumentation (the capability of this instrumentation is described in other areas of this SE). According to the FIP, the CNS SFP level instrumentation has a backup battery with sufficient capacity to ensure a minimum of 7 days of operation. Prior to the battery fully depleting, the licensee could utilize any 9 to 36 Vdc battery to supplement the installed backup to ensure indefinite SFP level monitoring capability.

3.3.5 Conclusions

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

3.4 Containment Function Strategies

The industry guidance document, NEI 12-06, Table 3-1, provides some examples of acceptable approaches for demonstrating the baseline capability of the containment strategies to effectively maintain containment functions during all phases of an ELAP event. One such approach is for a licensee to perform an analysis demonstrating that containment pressure control is not challenged.

The licensee performed a containment evaluation, NEDC 14-026, "MAAP Analysis to Support Cooper FLEX Strategy," which was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy of automatic containment isolation and venting the suppression pool through the hardened containment vent and concluded that the containment parameters of pressure and temperature remain well below the respective UFSAR

Section V-2, Table V-2-1, "Primary Containment System Principle Design Parameters and Characteristics," design limits of 56 psig and 281°F for more than 72 hours. From its review of the evaluation, the NRC staff noted that the required actions to maintain containment integrity and required instrumentation functions have been developed, and are summarized below.

Eventual containment cooling and depressurization to normal values may utilize off-site equipment and resources during Phase 3 if onsite capability is not restored.

3.4.1 Phase 1

In its FIP, the licensee stated that containment integrity is maintained by normal design features of the containment, such as the containment isolation valves and the HCVS. In accordance with NEI 12-06, the containment is assumed to be isolated following the event. As the torus heats up and the water begins to boil, the containment will heat up and pressurize. The HCVS is used to vent the torus with control from the control room. The licensee has performed a MAAP analysis (NEDC 14-026) to establish the containment venting control parameters during an ELAP. The analysis determined that containment should be vented at approximately 15 psig. Containment pressure should then be maintained between 5 psig and 15 psig until such time as another means of reducing torus temperature becomes available.

3.4.2 Phase 2

The Phase 2 strategy is to maintain the Phase 1 strategy of venting the torus through the HCVS. A portable FLEX generator is used to repower the instruments maintaining instrumentation used to monitor key containment parameters.

3.4.3 Phase 3

The FIP states that the Phase 2 strategy is continued throughout the event.

As part of the audit, the licensee clarified that the Phase 3 strategy for containment integrity is to continue the Phase 2 strategy with SAFER equipment providing defense-in-depth for onsite FLEX equipment (i.e. electrical power and pneumatics for the HCVS). This will maintain containment pressure and temperature within limits until the Phase 3 core cooling strategy as described in FIP Section 2.3.3 is implemented. Implementing the Phase 3 strategy for core cooling will maintain containment temperature within limits by controlling the heat input into containment and the HCVS will maintain containment venting capability to control pressure within limits.

3.4.4 Staff Evaluations

3.4.4.1 Availability of Structures, Systems, and Components

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

3.4.4.1.1 Plant SSCs

Containment

Section V.2 of the CNS UFSAR states that primary containment is a GE Mark I design pressure suppression system. The design employs a low-leakage pressure suppression containment system which houses the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor primary system. Primary containment consists of a drywell, a suppression chamber (torus) which stores a large volume of water (suppression pool), a connecting vent system between the drywell and the suppression pool, isolation valves, Primary Containment Isolation System, and vacuum relief system. Additional equipment including portions of ECCS is located within primary containment which provides services to primary containment. The primary containment is designed to withstand the SSE and the forces from any size breach of the reactor coolant pressure boundary up to and including an instantaneous circumferential break of the reactor recirculation piping. Primary containment has a maximum code allowable internal pressure of 62 psig.

The primary containment is completely enclosed within the reactor building. The reactor building is designed to withstand a SSE and is designed to withstand the impact of a tornado driven missile.

Hardened Containment Vent System (HCVS)

The FIP states that the HCVS is a severe accident capable, wetwell venting system that complies with the requirements of NRC Order EA-13-109. The flow path includes an MOV and two air operated valves (AOVs) in the line from the torus to the elevated release point on the reactor building roof.

The MOV, PC-MOV-233MV, is provided power during an ELAP from a dedicated PC233MV UPS. Shortly after the event occurs, a transfer switch is manipulated to provide the valve power from the UPS. The valve is opened and remains open for the rest of the event.

A separate HCVS UPS provides power to the controls for the air operated valves, as well as a number of indications and alarms. Four transfer switches are used to swap power to the HCVS UPS for all available components. The AOVs, PC-AOV-237AV and PC-AOV-AO32, have accumulators to provide sufficient pneumatic supply for operation for at least the first 24 hours of the event. A mechanical remote operating station (MROS) is installed to provide nitrogen as a pneumatic source for at least a week of valve operations

3.4.4.1.2 Plant Instrumentation

In NEI 12-06, Table 3-1, specifies that drywell pressure, drywell temperature, suppression pool temperature, and suppression pool water level are key containment parameters which should be monitored by repowering the appropriate instruments. In addition, valve indication for the HCVS will be monitored. The licensee's FIP states that control room instrumentation would be available due to the coping capability of the station batteries and associated inverters in Phase 1, or the portable DGs deployed in Phase 2. If no ac or dc power is available, the FIP states that key credited plant parameters, including these containment parameters, would be available using alternate methods. Emergency Procedure 5.3SBO, "Station Blackout," provides guidance for alternate method for monitoring key containment parameters.

3.4.4.2 Thermal-Hydraulic Analyses

The licensee performed NEDC 14-026, "MAAP Analysis to Support Cooper FLEX Strategy," to provide an analysis representing plant response during an ELAP. The analysis used the MAAP4 computer code. The analysis assumes a 10-inch hardened containment vent, RCS leakage of 66 gpm, which includes recirculation pump seal leakage of 18 gpm per pump for a total of 36 gpm, and technical specification unidentified RCS leakage of 30 gpm maximum. The calculation assumed the plant is operating at the current licensed power level of 2419 Mwt (megawatts thermal). In addition, the calculation conservatively assumed an initial containment pressure at 15.45 psia and torus venting initiated at 8-hours. The calculation determined, with the ECST tank unavailable, the peak torus pressure peaks at 35.1 per square inch absolute (psia) and decreases to less than 20 psia after the hardened vent is opened. By procedures 5.3SBO or 5.10FLEX30, venting starts at a drywell pressure of approximately 15 psig. The suppression pool temperature and drywell temperature peaks are 237°F and 271°F, respectively. The analysis was run for 24 hours. The output graphs show that by 24 hours, containment pressure and temperatures reach a steady state and remain below the design limits.

3.4.4.3 FLEX Pumps and Water Supplies

The FLEX pumps supporting containment integrity are used for removing decay heat from the reactor and rejecting it into the suppression pool. Containment pressure is maintained by venting the decay heat to the environment through the HCVS. Safety evaluation Section 3.2.3.5, "FLEX Pumps and Water Supplies," provides a complete description of the FLEX pumps and water supplies.

3.4.4.4 Electrical Analyses

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

The licensee's Phase 1 coping strategy is to maintain containment integrity using normal design features of the containment, such as the containment isolation valves and the HCVS. The installed HCVS has two dedicated UPSs, containing batteries, and battery chargers.

The dedicated HCVS UPS would supply power to the controls and indication for the two air operated valves in the HCVS flow path, PC-AOV-237AV and PC-AOV-AO32, as well as a number of indications and alarms. Plant operators would need to manipulate four transfer switches to swap power to the HCVS UPS for all required components. The HCVS UPS has a 24-hour capacity. Calculation NEDC 15-030, "HCVS UPS Sizing Analysis," Revision 1, showed that the 24-hour load requirement would be approximately 1 kVA (996 VA). The licensee selected a UPS that can supply 1.5 kilovolt Amperes (kVA) for 24 hours, which is adequate to supply the HCVS loads.

The second dedicated UPS, HCVS UPS-233, would supply power to a motor operated valve, PC-MOV-233MV. Plant operators would need to manipulate one transfer switch to swap power to the HCVS UPS-233. The HCVS UPS-233 has a 24-hour capacity. Calculation NEDC 15-

033, "HCVS PC233MV UPS Sizing Analysis," Revision 0, showed that the load requirement would be approximately 24 kVAMinutes. The licensee selected a UPS that can supply 125 kVAMinutes, which is adequate to supply the PC-MOV-233MV load.

During Phase 2, the licensee would maintain containment integrity by deploying a portable 60 kW FLEX DG to power the HCVS charger. The licensee would transition to Phase 2 prior to depleting the HCVS battery (i.e., within 24 hours). The licensee's analysis, NEDC 15-030, determined that during recharging of the UPS from a FLEX DG, the UPS input current during recharge (including the output loads) would be 42 amps at 120 Vac. This loading from the UPS during recharging is approximately 5 kW. Therefore, the 60 kW FLEX DG is adequate to supply the HCVS UPS load and the 25 hp well pump load of 28.3 kW for a combined load of approximately 33 kW. These two loads are the only loads on the FLEX DG at that time.

The licensee's Phase 3 strategy is to continue its Phase 2 strategy throughout the event. Cooper will receive offsite resources and equipment from an NSRC following 72 hours after the onset of an ELAP event. This equipment includes two 4160 Vac, 1 MW CTGs and a 480 Vac, 1100 kW CTG. Given the capacity of these generators, the NRC staff concludes that it is reasonable to expect that the licensee could utilize these resources to supply power to the HCVS components to maintain containment indefinitely.

Based on its review, the NRC staff determined that the electrical equipment available onsite (e.g., Class 1E batteries, HCVS UPSs, and a 60 kW FLEX DG), supplemented with the equipment that will be supplied from an NSRC (e.g., 4160 Vac CTGs and a 480 Vac CTG), provides sufficient capacity and capability to supply the required loads to maintain containment.

3.4.5 <u>Conclusions</u>

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

3.5 <u>Characterization of External Hazards</u>

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

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

The licensee reviewed the plant site against NEI 12-06 and determined that FLEX equipment should be protected from the following hazards: seismic; 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 NRC endorsement of

NEI 12-06 in JLD-ISG-2012-01. Guidance document NEI 12-06 directed licensees to proceed with evaluating external hazards based on currently available information. For most licensees, this meant that the OIP used the current design-basis information for hazard evaluation. Coincident with the issuance of Order EA-12-049, on March 12, 2012 (ADAMS Accession No. ML12053A340), the NRC staff issued a Request for Information pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) (hereafter referred to as the 50.54(f) letter), which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies. Due to the time needed to reevaluate the hazards, and for the NRC to review and approve them, the reevaluated hazards were generally not available until after the mitigation strategies had been developed. The NRC staff has developed a proposed rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was published for comment in the *Federal Register* on November 13, 2015, 80 FR 70610. The proposed MBDBE rule would make the intent of Orders EA-12-049 and EA-12-051 generically applicable to all present and future power reactor licensees, while also requiring that licensees consider the reevaluated hazard information developed in response to the 50.54(f) letter.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in response to the 50.54(f) letter and the requirements for Order EA-12-049 and the MBDBE rulemaking (COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards" (ADAMS Accession No. ML14309A256). The Commission provided guidance in an SRM to COMSECY-14-0037 (ADAMS Accession No. ML15089A236). The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 (ADAMS Accession No. ML15174A257), the NRC informed the licensees that the implementation of mitigation strategies should continue as described in licensee's OIPs, and that the NRC safety evaluations and inspections related to Order EA-12-049 will rely on the guidance provided in JLD-ISG-2012-01, Revision 0, and the related industry guidance in NEI 12-06, Revision 0. The hazard reevaluations may also identify issues to be entered into the licensee's corrective action program consistent with the OIPs submitted in accordance with Order EA-12-049.

As discussed above, licensees are reevaluating the site seismic and flood hazards as requested in the NRC's 50.54(f) letter. After the NRC staff approves the reevaluated hazards, licensees will use this information to perform flood and seismic mitigating strategies assessments (MSAs) per the guidance in NEI 12-06, Revision 2, Appendices G and H (ADAMS Accession No. ML16005A625). The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 (ADAMS Accession No. ML15357A163). The licensee's MSAs will evaluate the mitigating strategies described in this safety evaluation using the revised seismic and flood hazard information and, if necessary, make changes to the strategies or equipment. Licensees will submit the MSAs for NRC staff review.

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

3.5.1 Seismic

In its FIP, the licensee stated that the seismic criteria for CNS include two design-basis earthquake spectra; the operating basis earthquake at 0.1g and SSE at 0.2g. Per NEI 12-06 Section 5.2, all sites will consider the seismic hazard. It should be noted that the actual seismic hazard involves a spectral graph of the acceleration versus the frequency of the motion. Peak acceleration in a certain frequency range, such as the numbers above, is often used as a shortened way to describe the hazard.

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

3.5.2 Flooding

In its FIP, the licensee stated that the design-basis flood is the probable maximum flood (PMF) that has a value of 903 ft. MSL. The general ground elevation surrounding CNS Class I structures is elevated 13 ft. above the natural floodplain to 903 ft. MSL.

The finished floor elevation of all Class I structures is at elevation 903.5 ft. MSL, or 0.5 ft. above the PMF event. These structures were designed for a hydraulic load equivalent to a groundwater elevation of 903 ft. The station site grade level of 903 ft. MSL has been raised 13 ft. above the natural grade level of 890 ft. MSL, in order to bring final grade one foot above the existing 902 ft. MSL levee constructed by the U.S. Army Corps of Engineers (USACE). This levee was raised above its original design level and presently has a three foot minimum free board over the 1952 flood of record (899 ft. MSL). Flooding of the station is considered to be extremely unlikely due to the combination of upstream Missouri River flood control and the high final site grade. With respect to the 1,000 year, 10,000 year, and 1,000,000 year PMF floods, these water levels will provide 3.5 ft., 1.5 ft., and 0.5 ft. of freeboard respectively below the 903 ft. 6 in. grade floor elevation of the principle structures. The FSBs have a minimum floor elevation of 903.5 ft. MSL, which is equal to the finished floor elevation of all Class I structures, or 0.5 ft. above the PMF event specified in CNS UFSAR.

Regarding a PMP event, the licensee stated that Class I and Class II buildings are protected from the effects of precipitation through the use of roof drains and overflow scuppers. The remaining local site drainage is designed such that any excess rainfall not immediately absorbed into the ground will flow away from the buildings to be discharged into drywells or low lying areas adjacent to the plant site. Accordingly, these designs can safely remove the accumulated water from the probable maximum precipitation rate and can also accommodate the estimated 9.7 inch per hour (in./hr.) in one hour rainfall rate without adverse effects on the safety-related systems necessary for safe shutdown.

In its FIP, the licensee also stated that CNS does not credit any safety-related active ac powered dewatering systems for mitigating ground water intrusion into the portions of the plant which contain SSCs credited in the FLEX strategies or that require access for personnel during the BDBEE.

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

3.5.3 High Winds

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

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

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the design wind pressure for the station and structures is 30 lbs. per square foot which is the equivalent of sustained winds up to 100 mph. Station structures have been designed to withstand this wind velocity in accordance with American Society of Civil Engineers (ASCE) Paper 3269. Additionally, Class I structures are designed to the following:

Tornado design criteria:

- A tangential velocity of 300 mph
- A transverse velocity of 60 mph
- A pressure drop of 3 psi occurring over a 3 second time Interval

Tornado generated missile criteria:

- A 35 foot long utility pole with a 14 inch butt with an impact velocity of 200 mph
- A one-ton missile such as compact-type automobile with an impact velocity of 100 mph and a contact area of 25 square feet
- A 2 inch extra heavy pipe, 12 feet long
- Any other missile resulting from failure of a structure or component or one which has potential of being lifted from storage or working areas at the site. The CNS site is located in an area characterized by the NRC as having tornado design wind speeds greater than 130 mph and as such, the tornado hazard has been considered

Although the licensee did not address the impact of a hurricane in the integrated plan, the site is beyond the range of high winds from a hurricane per NEI 12-06 Figure 7-1. The NRC staff concludes that a hurricane hazard is not applicable and need not be addressed.

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

3.5.4 Snow, Ice, and Extreme Cold

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

In its FIP, the licensee stated that the design low outside temperature is -5°F dry bulb which will only be exceeded 1 percent of the time during the winter. The CNS site is located within the region characterized by the National Oceanic and Atmospheric Administration (NOAA) as having a 3-day snowfall of up to 18 inches and would need to consider snow removal in the deployment of the FLEX strategy. The CNS site is also located within the region characterized by EPRI as ice severity level 4. As such, the CNS site is subject to severe damage to power lines and/or existence of large amounts of ice.

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 its FIP, the licensee stated that all sites will address high temperatures and as such, the CNS design high outside temperature is 97°F dry bulb (79°F wet bulb). Based on historical records, this temperature is only expected to be exceeded 1 percent of the time during the summer.

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

3.5.6 Conclusions

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

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

In its FIP, the licensee stated that the FLEX equipment is stored in FSBs, one located inside the protected area in close proximity to the fire pump house and the second outside the protected area at the southwest corner of the low level radioactive water (LLRW) storage pad. The structures are constructed of standard light gauge metal having 6-20 ft. long international standards organization containers in combination with two additional vehicle bays constructed of light gauge metal.

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

3.6.1.1 Seismic

In its FIP, the licensee stated that the FSBs are designed utilizing NEI 12-06, Section 5.3.1 configuration b, in accordance with the seismic requirements of ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures." During subsequent reviews of the FSB design, it was determined that the buildings were designed using values less than the SSE. Subsequently, the licensee completed an evaluation which determined that other loadings on the buildings were sufficient to bound the loadings that would be experienced during an SSE and the buildings would remain operational following the SSE. Tie down points are provided inside the FSBs to secure equipment and protect the equipment from interacting with each other in the event of an earthquake.

3.6.1.2 Flooding

In its FIP, the licensee stated that the protection of FLEX equipment against external flooding events is performed in accordance with NEI 12-06, Section 6.2.3.1.a, which states that equipment is protected from floods if it is stored above the flood elevation from the most recent site flood analysis. The design-basis flood level at CNS is 903 ft. MSL. The finished floor elevation of the FSB slabs is at 903 ft. 6 in. to ensure protection against a PMF consistent with all other major site buildings and facilities.

3.6.1.3 High Winds

In its FIP, the licensee stated that CNS is located in high wind hazard (tornado) Region 1 with anticipated wind speeds greater than 200 mph (NEI 12-06 Figure 7-2). Cooper's FSBs are designed utilizing NEI 12-06, Section 7.3.1.1, configuration b, and in accordance with the wind load requirements of ASCE 7-10. The licensee compared the CNS UFSAR description of a tornado path and size to a statistical analysis of all tornados reported by the National Weather Service Storm Prediction Center from 1950 to 2015.

Subsets of the NOAA Severe Weather GIS Data were chosen for evaluation:

- Continental United States
- 400 mile radius originating at CNS
- Four state region consisting of Nebraska, Iowa, Kansas, and Missouri

- 250 mile radius originating at CNS
- 100 mile radius originating at CNS

No notable differences were apparent in any of the data sets that would impact the evaluation. The predominant vector of tornadoes within 250 miles of CNS is 22.2 degrees south of east. There are also a significant portion of tornados that approach directly from the southwest. Larger tornados move more west southwest to east northeast. As two trajectories are prominent in the subsets, both were considered applicable for the evaluation.

These trajectories are:

- Directly along the southwest to northeast line
- 22.2 degrees south of east line

Based on the analysis, tornado size distribution shows that 90 percent of all tornados in the data set are 600 ft. in width. Tornado size was conservatively taken as the UFSAR defined tornado with a width of 750 ft., based on the trajectories evaluated. The licensee determined that the minimum separation distance that exists between the facilities is 869 ft. which is larger than the required distance. The FSBs are located in diverse locations; one of the buildings is located next to robust structures including the radwaste building, condensate storage tanks, fire protection tanks, control building, and to some extent the reactor building. The other FSB is located approximately halfway between the Missouri River and the western bluffs. Both buildings are identical in design and utilize a reinforced concrete slab to support and retain all structures from postulated environmental conditions.

3.6.1.4 <u>Snow, Ice, Extreme Cold and Extreme Heat</u>

In its FIP, the licensee stated that the protection of equipment from impacts due to extreme high temperatures is performed in accordance with NEI 12-06 Section 9.3.1, which states that equipment should be maintained at a temperature within a range to ensure its likely function when called upon. The FSBs ventilation system is designed to limit high temperatures within the storage spaces to within 6°F of outside ambient temperature using power ventilators. An electric unit heater rating of 2 kW maintains the subject building storage areas at a minimum indoor design temperature of 60°F on a design-basis winter day with an ambient temperature of -30°F. A fire detection system is installed in each of the storage buildings. The systems are designed and installed per the latest edition of NFPA 70, "National Electrical Code," and NFPA 72, "National Fire Alarm Code."

3.6.1.5 <u>Conclusions</u>

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

3.6.2 <u>Availability of FLEX Equipment</u>

Section 3.2.2.16 of NEI 12-06 states, in part, that in order to assure reliability and availability of the FLEX equipment, the site should have sufficient equipment to address all functions at all units onsite, 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 concludes that, if implemented appropriately, the licensee's FLEX strategies include a sufficient number of portable FLEX pumps, FLEX DGs, and equipment for RPV makeup and core cooling, SFP makeup, and maintaining containment consistent with the N+1 recommendation in Section 3.2.2.16 of NEI 12-06.

3.7 Planned Deployment of FLEX Equipment

In its FIP, the licensee stated that the pre-determined haul paths have been identified and documented in the FSGs. The haul paths attempt to avoid areas with trees, power lines, narrow passages, etc. when practical. However, high winds can cause debris from distant sources to interfere with planned haul paths. The licensee documented the acceptability of the location of the FSBs from a tornado/missile hazard standpoint and found them to be acceptable from a diverse location standpoint such that the site's FLEX capability will remain functional following a high wind/tornado event. The rationale is that there is sufficient separation between the FSBs such that the event can only affect one. The FSBs each contain a smaller tractor/loader that can move all debris likely to be generated onsite except for vehicles. The same rationale applies to the debris that the loader is needed to remove, i.e. vehicles. The vehicles that the loader would have to move are all located in the parking lots outside of the protected area between LLRW pad and the FLEX deployment area on the north side of the plant. For high wind/tornado events that affect the FSB on the LLRW pad, the FSB on the north side of the plant will be unaffected, clear of large debris and not require the loader to be used. For high wind/tornado events that affect the FSB on the north side of the plant, the FSB and loader on the LLRW pad will be unaffected and able to clear the debris. Therefore, at least one piece of equipment remains functional and deployable to clear obstructions from the pathway between the FSBs and its deployment location(s).

In its FIP, the licensee stated that the potential impairments to required access are doors and gates, and site debris blocking personnel or equipment access. The coping strategy to maintain site accessibility through doors and gates is applicable to all phases of the FLEX coping strategies, but is immediately required as part of activities required during Phase 1. Doors and gates serve a variety of barrier functions on the site. One primary function is security. However, other barrier functions include fire, flood, radiation, ventilation, and tornado. As barriers, these doors and gates are typically administratively controlled to maintain their function as barriers during normal operations. Following a BDBEE and subsequent ELAP event, FLEX coping strategies require the routing of hoses and cables through various barriers in order to connect FLEX equipment to station fluid and electric systems. For this reason, certain barriers (gates and doors) are opened and remain open. This violation of normal administrative controls is acknowledged and is acceptable during implementation of FLEX coping strategies. Non-licensed operators have a key ring on their person which allows access to all areas of the plant. Licensed operators may obtain a key ring from Security. The ability to open doors for ingress and egress, ventilation, or routing of temporary cables/hoses is necessary to implement the FLEX coping strategies. The deployment of onsite FLEX equipment to implement coping strategies beyond the initial plant capabilities (Phase 1) requires that pathways between the

FSBs and various deployment locations be clear of debris resulting from beyond-design-basis (BDB) seismic, high wind (tornado) or flooding events.

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

3.7.1 <u>Means of Deployment</u>

Debris removal equipment is stored inside the FSBs to be protected from the severe storm and high wind hazards. Additionally, CNS obtained a front end loader for removing large debris in the event of a beyond-design-basis external event. The loader was originally planned to be stored inside the turbine building. However, the licensee determined that vehicles should not be stored permanently in that area without installed fire suppression. Accordingly, the LLRW pad near the FSB has been determined to be an acceptable location.

Cooper is in an area that experiences snow and ice during the winter and, as such, develops a snow removal plan each season. The FSBs each contain a small tractor/loader that is capable of removing any snow or ice accumulation if the site's removal equipment is unavailable. Additionally, each FSB contains a tractor that can be used to tow FLEX pumps, DGs, refueling equipment and air compressors.

3.7.2 <u>Deployment Strategies</u>

The licensee evaluated potential soil liquefaction from a severe seismic event. The deployment path covers two unique soil types. The first is in CNS structural fill. The second is in native alluvium that surrounds the CNS structural fill. The FSB inside the protected area at CNS is in type II structural fill which is not expected to liquefy. At CNS, type II structural fill covers underlying Alluvium, which is expected to liquefy. One facility, and the two deployment paths from this facility, are built in structural fill. The expected settlements are minimal, which range from 0.47 in. to 1.17 in. The licensee estimated that the settlement due to liquefaction of the native alluvium to be between about 4 in. and 14 in. following the review level earthquake (RLE). The median settlement is 8.7 in. with a mean of 8.7 in. The RLE is 0.3g which is greater than the SSE of 0.2g. This indicates that the expected settlements will be on the lower end of the estimates.

Furthermore, the FLEX equipment includes earthmoving equipment that have the capacity to handle the settlements described above. Three conclusions were made in relation to liquefaction and CNSs FLEX portable equipment deployment paths:

- Portions of the deployment path in structural fill will produce minimal settlements.
- Portions of the deployment path not in structural fill will experience settlements averaging 8.7 inches.
- CNS has adequate measures in place to cope with the expected settlements produced from soil liquefaction during severe seismic events. These measures include diverse deployment pathways, rugged equipment selection, and a FLEX strategy including earthwork equipment.

Therefore, potential soil liquefaction from the severe seismic event will not impede FLEX equipment deployment.

In its FIP, the licensee stated that in the event that the backup source of water (Missouri River) would need to be used, any ice that forms along the bank can be removed by the site's debris removal equipment. Generally speaking, along rivers and streams, ice formation takes place as frazil ice in the center of a stream and shore ice growth along the borders of the stream. During the winter, USACE will closely monitor ice conditions below Garrison, Oahe and Gavins Point dams and make reservoir regulation adjustments to lessen the impact of river ice formation.

In its FIP, the licensee stated that for the core cooling strategy prior to transitioning to Phase 2, FLEX 175 kW and FLEX 60 kW DGs are staged near the optimum water chemistry (OWC) building. The FLEX 175 kW DG is deployed approximately 2 hours into the BDBEE and is operational, supporting battery charger operation, around 5 hours into the event. The FLEX 60 kW DG is deployed approximately 2 hours into the BDBEE and is operational around 8 hours into the event.

The licensee also described an alternate core cooling strategy. The North well can provide makeup to the vessel via FLEX pump with hose routed to either the RHRSW FLEX connection or the RHR 'B' B.5.b connection. The makeup water is then injected to the RPV through the RHR 'A' or 'B' injection lines. Hose from the North well is routed south to the laydown area west of the multi-purpose facility (MPF). A 20,000 gallon bladder can also be staged in this area to be used as a surge volume, if required, and can receive the discharge from the north well pump. The bladder or the North well pump will provide a suction source for a FLEX pump, also staged in the laydown area.

The licensee stated that for containment pressure control, the HCVS is utilized to maintain containment at 5 to 15 psig thereby ensuring containment integrity. FLEX equipment is staged to provide a power supply to the HCVS UPS / battery charger, this in turn will maintain the ability to open and close the HCVS containment vent valve during the event. The FLEX 60 kW DG, as discussed above, will be used as the power source for the HCVS battery charger.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling

In the FIP, Sections 2.3.4.5 and 2.3.4.6 describe the primary and alternate core cooling connection points for the portable FLEX pump. The FLEX pump will supply water to the connection points via hose. The primary connection point is at a FLEX connection on the combined RHRSW crosstie to the RHR injection flow path (upstream of SW-V-120). Water is injected through the emergency core flooding cross-tie to the RHR A subsystem. The alternate connection point is the RHR B subsystem, at a B.5.b connection. Water is injected through the RHR B injection line. The primary connection is located in the control building and the alternate connection is located in the reactor building. Both buildings are Class I structures designed to be robust with respect to a SSE, wind loading, tornado loading, and tornado generated missiles.

In the FIP, Section 2.9.2 describes hose routings and staging locations. Well water will be supplied to the FLEX pump either directly from the North well pump or through a 20,000 gallon bladder that can be used as a surge volume, if necessary. A hose will be routed south from the North well to the laydown area west of the MPF. The FLEX pump will also be staged in the laydown area. For the primary discharge connection, a hose will be routed from the FLEX pump to the RHRSW FLEX connection. For the alternate discharge connection, a hose will be routed to the RHR B, B.5.b connection.

SFP Cooling

In the FIP, Sections 2.4.4.1 and 2.4.4.2 describe primary and alternate SFP connections. Similar to core cooling, the portable FLEX pump is also used for SFP cooling. Its source of water is the North well pump or bladder as described previously for core cooling. The FLEX pump's primary connection to the SFP is through the FPC system chemical decontamination connection. Valves are aligned to supply water through this system into the SFP through a normal fill location. For the alternate strategy, a hose is run from the discharge of the FLEX pump directly to the SFP. No physical connections to permanent plant equipment are required.

3.7.3.2 Electrical Connection Points

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

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

The deployment area of the 175 kW FLEX DG is near the north side of the turbine building. The licensee installed two fusible disconnect switches to allow the 175 kW FLEX DG to provide power to the 125 Vdc battery charger C and 250 Vdc battery charger C. These disconnects allow for separation of the connection points from the Class 1E electrical system during normal operation. The connection points provide the ability to charge either Division 1 or Division 2, 125 Vdc and 250 Vdc batteries and supply dc loads. As discussed above, the licensee would route color-coded cables from the FLEX 175 kW FLEX DG to the "C" battery charger room.

As an alternate strategy to power the batteries from the 'C' chargers, the licensee could use pre-staged cables to connect the 175 kW FLEX DG directly to the feeder breakers for the 'A' or 'B' 125/250 Vdc battery chargers on MCC LX or TX. The FSGs 5.10FLEX.01, 5.10FLEX.02, 5.10FLEX.03, and 5.10FLEX.04 include guidance for staging, cable routing, connecting, and operating the 175 kW FLEX DGs.

A 120 VAC power cord could also be routed from the 175 kW FLEX DG to the 120 Vac vital instrumentation receptacle located in the cable spreading room. The power cord would power vital level instrumentation located in the control room. Procedure 5.10FLEX.28, "Vital Instrumentation FLEX Operations," Revision 0, contains guidance for using a 175 kW FLEX DG (120 Vac power) or the Phase 3 4160 Vac CTGs (through either Bus F or G) to supply vital instrumentation in the control room.

The function of the 60 kW FLEX DG is to supply power for makeup water sources and to recharge the battery for the HCVS after 24 hours. The makeup water sources are the main condenser hotwell (a one-time transfer) and the North well (long-term make up). The licensee would stage a 60 kW FLEX DG on the north side of the OWC building and turbine building. Procedure FSG 5.10FLEX.05, "Reliable Hardened Containment Vent Battery Charger Tie-In," Revision 0, includes guidance for staging, cable routing, connecting, and operating the 60 kW FLEX DG. In its FIP, the licensee stated that the correct phase rotation of the 60 kW FLEX DG was verified during modification acceptance testing.

For Phase 3, procedures 5.10FLEX.05 and 5.10FLEX.10, "ECST Makeup from North Well," Revision 1, includes guidance for using the NSRC 480 Vac CTG as a backup for the Phase 2 FLEX DGs supplying the Class 1E battery charger, make-up pump, and HCVS battery charger. According to the licensee, the hotwell has limited capacity and would not be refilled, so it would not be operated when Phase 3 equipment is provided. Therefore, instructions for utilizing Phase 3 equipment is not required for the hotwell pump. Cooper does require a conversion cord to go from 4/0 AWG to #2 AWG cables. Cooper maintains 4 sets (1 set for each phase color) in each FSB. Procedure 5.10FLEX.10 contains guidance for verifying proper phase rotation of the make-up pump.

Procedures 5.10FLEX.07 and 5.10FLEX.08 include guidance for cable routing, connecting, and operating the 4160 kW FLEX CTGs. Procedure 5.10FLEX.07 and 5.10FLEX.08 include guidance for verifying proper phase rotation prior to energizing required equipment.

3.7.4 Accessibility and Lighting

In its FIP, the licensee stated that the battery powered emergency lighting is available throughout the plant, including the control room, battery room, DG rooms, critical service switchgear areas, stairways, and exits. This lighting will only be available for a short time (1.5 to 8 hours) into an event. Portable dc and ac lighting is stored in the FSBs. Three large battery powered portable area lights are stored in the FSB. One will be used in the control room, one at the portable DG deployment area and one at the well area. These lights have a 24 hour capacity (LED). After 24 hours, a portable DG will supply ac lighting to the control room and additionally be able to recharge the battery powered lights during daylight hours. Operators will have flashlights for use in areas where other lighting is minimal. Sufficient replacement batteries are stored in the FSBs.

3.7.5 Access to Protected and Vital Areas

Vehicle access to the protected area is via the double gated sallyport at the security building. As part of the security access contingency, the sally-port gates are manually controlled to allow delivery of FLEX equipment (e.g., generators, pumps) and other vehicles such as debris removal equipment into the protected area.

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

3.7.6 Fueling of FLEX Equipment

In its FIP, the licensee described the FLEX strategies to supply fuel to necessary diesel power generators, pumps hauling vehicles, etc. The general coping strategy for supplying fuel oil to diesel-driven portable equipment (i.e., pumps and generators), being utilized to cope with an ELAP, is to draw fuel oil out of any available existing diesel fuel oil tanks on the CNS site. The primary source of fuel oil for portable equipment is two bunkered, seismic Class 1 fuel oil storage tanks with two additional day tanks located in seismic Class I structures. All the tanks are protected from winds, floods, tornados and missiles. These tanks will be the initial supply.

All together, these tanks contain 52,500 gallons of available fuel oil. The two storage tanks are buried and their appendages are protected by a substantial cover. The manholes providing access to the capped fill connections and the tank vents are all located above 906 ft. MSL. Based on published ratings for the FLEX equipment, the EDG fuel oil storage tanks have a capacity of greater than 30 days. The quality of fuel oil in EDG fuel oil storage tanks is maintained in accordance with the diesel fuel oil testing program described in the technical specifications. Fuel oil in the fuel tanks of portable diesel engine driven FLEX equipment is maintained in the preventative maintenance program in accordance with the manufacturer's guidance and existing site maintenance practices. Fuel oil is transported to FLEX equipment in a 100 gallon fuel tank mounted on a portable trailer. This self-contained trailer contains a portable generator, electric pump for pumping fuel into the 100 gallon tank, and an attached fuel pump for pumping the contents of the 100 gallon tank to the FLEX equipment. There are two trailers, one stored in each FSB. The trailer is towed by one of two tractors, one stored in each FSB.

3.7.7 Conclusions

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

3.8 Considerations in Using Offsite Resources

3.8.1 Cooper Nuclear Station SAFER Plan

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

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

In its FIP, the licensee described that in the event of a BDBEE and subsequent ELAP condition, equipment is moved from an NSRC to a local assembly area established by the SAFER team. The SAFER response plan for CNS has been approved. Staging areas, travel paths and congested area flight plans have all been walked down and approved by NSRC personnel.

The NRC staff noted that the licensee's SAFER response plan contains (1) SAFER control center procedures, (2) NSRC procedures, (3) logistics and transportation procedures, (4) staging area procedures, which include travel routes between staging areas to the site, (5) guidance for site interface procedure development, and (6) a listing of site-specific equipment (generic and non-generic) to be deployed for FLEX Phase 3.

By letter dated September 26, 2014 (ADAMS Accession No. ML14265A107), the NRC staff issued its assessment of the NSRCs established in response to Order EA-12-049. In its assessment, the staff concluded that SAFER has procured equipment, implemented appropriate processes to maintain the equipment, and developed plans to deliver the equipment needed to support site responses to BDBEEs, consistent with NEI 12-06 guidance; therefore, the staff concluded in its assessment that licensees can reference the SAFER program and implement their SAFER response plans to meet the Phase 3 requirements of Order EA-12-049.

3.8.2 Staging Areas

In general, up to four staging areas for NSRC supplied Phase 3 equipment are identified in the SAFER plans for each reactor site. These are a Primary (Area C) and an Alternate (Area D), if available, which are offsite areas (within about 25 miles of the plant) utilized for receipt of ground transported or airlifted equipment from the NSRCs. From Staging Areas C and/or D, the SAFER team will transport the Phase 3 equipment to the on-site Staging Area B for interim staging prior to it being transported to the final location in the plant (Staging Area A) for use in Phase 3. For CNS, Staging Area C is the Tecumseh Municipal Airport and Staging Area D is the Nebraska City Municipal Airport. From these sites, equipment can be taken to the CNS site to Staging Area B in the southwest parking lot. Staging Area B is accessible by helicopter if ground transportation is unavailable. Cooper has a Memorandum of Understanding with the Nebraska Emergency Management Agency for support during Phases 2 and 3 of FLEX. Communications are established between the site and the SAFER team via satellite phones and required equipment moved to the site as needed. First arriving equipment is delivered to the site within 24 hours from the initial request. The order at which equipment is delivered is identified in the CNS's SAFER Response Plan.

3.8.3 Conclusions

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

3.9 Habitability and Operations

3.9.1 Equipment Operating Conditions

3.9.1.1 Loss of Ventilation and Cooling

Following a BDBEE and subsequent ELAP event at CNS, ventilation that provides cooling to occupied areas and areas containing required equipment will be lost. The primary concern with regard to ventilation is the heat buildup that occurs with the loss of forced ventilation in areas that continue to have heat loads.

The licensee performed loss of ventilation analyses to quantify the maximum steady-state temperatures expected in specific areas related to FLEX implementation. The analyses ensure that the environmental conditions remain acceptable for personnel habitability or accessibility and within equipment limits. The key areas identified for all phases of execution of the FLEX strategy activities are the 932 ft. elevation of the control building (control room), 903 ft. elevation of the control building (battery and switchgear rooms), RCIC quad and steam tunnel, RHR pump room, containment (ADS), and areas containing required instrumentation.

932 ft. Elevation of the Control Building (Control Room)

During an ELAP, some control room vital electronics, instrumentation and emergency lighting remain energized from emergency dc power sources. Licensee calculation NEDC 15-002, "Review of Tetra Tech Portable Equipment Calculations in support of CNS FLEX Strategy," Revision 1, contains the licensee's evaluation of loss of ventilation as a result of an ELAP. The licensee's evaluation showed that during an ELAP, assuming a 97°F outside air temperature with no night time relief, temperatures in the 932 ft elevation control building control room would be slightly above 110°F after approximately 40 hours. The peak temperature at 72 hours would be approximately 112°F.

The calculation showed that by implementing compensatory measures in the form of flexible ducting on a 40 in. (~11,000 cubic feet per minute (cfm)) portable fan from the outside through various doors 8 hours into an ELAP event, the high temperatures would be rapidly mitigated. With the prescribed compensatory measures, the peak temperature for the control room is acceptable for personnel and equipment. 5.10FLEX.19, "Alternate Ventilation FLEX Operations," Revision 0, contains guidance for establishing portable ventilation during an ELAP.

Based on the licensee's analysis and the availability of procedures to maintain temperatures below 120°F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, for electronic equipment to be able to survive indefinitely), the NRC staff concludes that the electrical equipment in the control room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

903 ft. Elevation of the Control Building (Battery and Switchgear Rooms)

According to the base case simulation in licensee calculation NEDC 15-002 for the 903 ft. elevation of the control building, where the battery rooms and switchgear rooms are located, temperatures would be less than 120°F for all rooms up to 6 hours prior to heat loads from the A, B, or C chargers coming online during Phase 2. Within 30 minutes of an ELAP event, operators would open select doors using guidance contained in Emergency Procedure 5.3SBO.

The licensee's analysis shows that beyond 6 hours with battery charging heat loads coming online, medium (~4,000 cfm) fans are directed into the battery charger rooms. However, only rooms with an operating charger will require a fan (Note: no more than two chargers will run concurrently). Additionally, in order to manage the temperature in controlled corridors outside of these rooms, a medium (~4,000 cfm) fan would need to be placed in a door from the turbine building.

Based on the results of the licensee's evaluation, operators would deploy three portable ventilation fans that would be powered by a FLEX 175 kW FLEX DG during Phase 2 to ventilate the battery and switchgear rooms. This strategy would provide ventilation for heat removal as well as for hydrogen gas removal. This exhaust path is the same path that is used for the licensee's current station blackout strategy. The portable fans' capacity is greater than the plant's normal ventilation capability. The FSGs 5.10FLEX.01, 5.10FLEX.02, 5.10FLEX.03, and 5.10FLEX.04 direct operators to open doors and establish portable ventilation for the 903 ft elevation of the control building.

The licensee also performed multiple sensitivity studies including the above case that confirmed that all rooms on the 903 ft. elevation of the control building containing equipment required for the licensee's mitigating strategies remain below their design limits during an ELAP event.

Elevated temperatures also have an impact by increasing the charging current required to maintain the float charging voltage set by the battery charger. The elevated charging current will in turn increase cell water loss through an increase in gassing. Based on this, periodic water addition may be required or the float charging voltage reduced per the guidance contained in the C&D Technologies vendor manual.

Based on the licensee's analysis and the availability of procedures to maintain temperatures below 120°F (the temperature limit in NUMARC-87-00 for electronic equipment to be able to survive indefinitely and a temperature that is below the battery manufacturer limits), the NRC staff concludes that the electrical equipment in the 903 ft. elevation of the control building (battery and switchgear rooms) should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

RCIC Quad and Steam Tunnel

The RCIC room will have a continuous heat load under ELAP conditions in Phases 1 and 2, since RCIC is utilized throughout the event as the primary source of core cooling. The RCIC system is designed for continuous operation at a temperature of 148°F and 100 percent relative humidity. Licensee calculation NEDC 15-002 determined that opening and closing doors is not sufficient to preclude exceeding 148°F in the RCIC quad. Licensee calculation NEDC 15-002 determined that a small (~1,500 cfm) fan and ducting would effectively displace the hot air in the room and turn it over with cooler air from a cooler location. The model demonstrates required deployment of this fan is needed between 8 and 12 hours. Additionally, the licensee opens doors and roof hatches within 3 hours to promote natural circulation. The model demonstrates that opening these doors at 4 hours is acceptable. Procedure FSG 5.10FLEX.19 provides guidance for establishing ventilation in the RCIC quad. Procedure 5.3SBO contains a flow chart that would direct operators to implement 5.10FLEX.19 within the required times.

According to NEDC 15-002, the temperature in the steam tunnel peaks at just above 275°F after 5 minutes and then decays away with the decreased heat load from the steam lines (due to cooldown). The operability limit in the steam tunnel is 308°F for RCIC-MOV-MO16. Therefore, the steam tunnel does not require any portable ventilation or doors to be opened during an ELAP event.

The licensee also performed multiple sensitivity studies including the above case that confirmed that the RCIC quad and the steam tunnel remain below the design limits of required equipment during an ELAP event. Additionally, CNS will receive offsite resources and equipment from an NSRC following 72 hours after the onset of an ELAP event. The licensee's Phase 3 activities should further reduce the heat load in the reactor building by either placing suppression pool cooling or shutdown cooling in service.

Based on temperatures remaining below the design limits, the NRC staff concludes that the electrical equipment in the RCIC quad and the steam tunnel will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

RHR Pump Room

During Phase 3, the licensee plans to place an RHR pump into service in order to perform shutdown cooling and suppression pool cooling. This would result in heat addition to the RHR pump room due to heat generated by the RHR pump motor as well as heat dissipated from the associated piping and RHR heat exchanger. For long-term RHR pump operation, the RHR pump room should be cooled to maintain room temperatures within acceptable ranges (limited by maximum allowable RHR pump motor requirements). Once the NSRC supplied 4160 Vac CTGs are connected to a 4160 Vac bus, operators would restore power to reactor equipment cooling and the RHR room cooler. Reactor equipment cooling is used to supply flow to the RHR pump bearing and seal oil coolers, as well as the RHR pump room fan cooling unit. FSGs 5.10FLEX.23, "Reactor Equipment Cooling FLEX Operations," Revision 0, 5.10FLEX.07, and 5.10FLEX.08 provide guidance for restoring power to reactor equipment cooling and the RHR room cooler.

Based on the restoration of the cooling systems that normally support the RHR pump operation, the NRC staff concludes that the electrical equipment in the RHR pump room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Containment (Automatic Depressurization System)

Calculation NEDC 14-026, and ERIN Calculation C122140001-11622, "MAAP Analysis to Support Cooper FLEX Strategy," Revision 1, showed that the drywell airspace temperature will reach steady state at approximately 250°F at 24 hours. This temperature will maintain for about 72 hours when shutdown cooling is implemented. This temperature is below the maximum temperature of 340°F at which the ADS is qualified. Additionally, the target rock solenoid valves (safety relief valves) peak and continuous test temperatures are 355°F (maximum) and 265°F (100 days), respectively.

Cooper will start receiving offsite resources and equipment from an NSRC following 24 hours after the onset of an ELAP event. The licensee could utilize these resources to reduce or maintain temperatures within primary containment to ensure that required electrical equipment survives indefinitely, if necessary.

Based on temperatures remaining below the design limits of equipment, the NRC staff concludes that the electrical equipment in containment should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Areas Containing Required Instrumentation

For required FLEX instrumentation in the control building, licensee calculation NEDC 15-002 showed that area temperatures are steady-state and would remain less than 120°F. For the ECST room elevation 877 ft. 6 in., the peak temperature would reach the design temperature of 131°F and the steady-state temperature would be 100°F. For the control building corridor elevation 903 ft. 6 in., the peak temperature would reach 108.4°F and the steady-state temperature would be 108.4°F. For the cable spreading room elevation 918 ft., the peak temperature would reach 110.9°F and the steady-state temperature would be 110.9°F. For the control room elevation 932 ft. 6 in., the peak temperature would reach 111.7°F and the steady-state temperature with a diurnal cycle would be 111.7°F. Based on the temperatures remaining below 120°F (the temperature limit, as identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, for electronic equipment to be able to survive indefinitely), the NRC staff concludes that the required instrumentation in the control building should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

For electrical instrumentation that is environmentally qualified (EQ), the peak EQ temperatures for the areas with required FLEX instrumentation are above the predicted area temperatures during an ELAP event (NEDC 15-002) for the short-term. For the general area of the reactor building elevation 903 ft. 6 in., the peak EQ temperature is 253°F compared to a peak ELAP temperature of 137.3°F. For the general area of the reactor building elevation 931 ft. 6 in., the peak EQ temperature is 211°F compared to a peak FLEX temperature of 128.8°F. For the ASD room in the reactor building elevation 913 ft. 6 in., the peak EQ temperature would reach 138°F. While the licensee did not specifically model the ASD room, the peak temperature should remain below the peak EQ temperature of 138°F based on the fact that the ASD room does not interact with the general reactor building elevation 903 ft. 6 in. airspace and no significant heat loads exist in the ASD room. Furthermore, the door to the ASD room is normally closed, effectively separating and insulating the ASD instrumentation. For the long-term, when the railroad airlock door is sufficiently cleared and opened, the steady-state area temperatures in the reactor building improve significantly. For the general area of the reactor building elevation 903 ft. 6 in., the EQ temperature at 180 days is 133°F compared to a long-term steady-state ELAP temperature of 114°F. For the general area of the reactor building elevation 931 ft. 6 in., the EQ temperature at 180 days is 129°F compared to a long-term steady-state ELAP temperature of 112.6°F. For the ASD room in the reactor building elevation 913 ft. 6 in., the EQ temperature at 180 days is 133°F. The general area of the reactor building elevation 903 ft. 6 in. is the primary heat load into the ASD room. With the railroad airlock opened, this heat load would be mitigated and the ASD room temperature would respond accordingly. Based on the above, the NRC staff concludes that the required FLEX instrumentation that is environmentally qualified should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

For the remaining non-EQ electrical and mechanical FLEX instrumentation in the reactor building, maximum ambient operating temperatures were verified to be acceptable through vendor guidance or documentation. Based on this material, the instrumentation is rated for a maximum ambient operating temperature of 150°F. In all FLEX calculation cases, the short-term peak temperature and steady-state long-term temperatures during an ELAP would be less than 150°F. Two required FLEX instruments did not meet the temperature requirements. Instrument NBI-LI-185B provides wide range level indication at the high-pressure coolant injection (HPCI) ASD panel in the ASD Room. Instrument NBI-LI-185B is a Yarway 4459. Similar 4000 series Yarways carry an EQ rating. Based on this similar design from the same manufacturer and the peak FLEX temperatures and steady-state temperatures being significantly less than the EQ temperature profile temperatures, the NBI-LI-185B should remain functional for the duration of the ELAP event with no adverse impact to measurement of wide range level. For PC-TI-2A, C, E, and G in the ASD Room, the maximum operating temperature is 122°F. Based on the lack of interaction and the insulation between the ASD room and the heat loads in the general area of the reactor building, the ASD room temperature is not expected to be greater than 122°F. It is possible that the ambient temperature could increase to just above 122°F if the railroad airlock door is blocked by debris in the short-term. Brief exposure to ambient temperatures within 10 percent of the maximum rating will not adversely impact the instrumentation based on margin built into the design by the manufacturer. In the unlikely event the PC-TI-2A, C, E, and G becomes non-functional, PC-TE-1A-H and PC-TE-2A-H will provide torus temperature indication in place of PC-TI-2A, C, E, and G in the ASD room in the reactor building. PC-TE-1A-H and PC-TE-2A-H are in the cable spreading room in the control building and the FLEX analysis (NEDC 15-002) shows that the cable spreading room area temperatures are below 120°F indefinitely. Based on the above, the NRC staff concludes that the required FLEX instrumentation that is non-EQ should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Based on its review of the essential station equipment required to support the FLEX mitigation strategy, which are primarily located in the 932 ft. elevation of the control building (control room), 903 ft. elevation of the control building (battery and switchgear rooms), RCIC quad and steam tunnel, RHR pump room, containment (ADS), and areas containing required instrumentation, the NRC staff concludes that the equipment should perform their required functions at the expected temperatures as a result of loss of ventilation during an ELAP event.

3.9.1.2 Loss of Heating

FLEX equipment is stored in one of two FSBs. In the FIP, Sections 2.7 and 2.11.2 state that the storage areas within these buildings are maintained at a minimum indoor design temperature of 60 °F for an ambient temperature of -30°F on a design-basis winter day. As stated in FIP Section 2.11.2, the licensee's FLEX strategy does not credit heat tracing. Section 8.3.1 of NEI 12-06 states that FLEX equipment should be maintained at a temperature within a range to ensure it is likely to function when called upon. The NRC staff concludes that by maintaining temperature in the FSB storage areas to at least 60°F, the FLEX equipment's low temperature specification would not be reached, and therefore, the equipment should be available to function when called upon consistent with Section 8.3.1 of NEI 12-06.

Other equipment should also be available when required. Hoses that are routed outside and pumps that are located in the laydown area would have positive flow. With positive flow, the hoses and pumps would be protected from freezing. Therefore, the NRC staff concludes that equipment such as pumps and hoses should also be available to support the FLEX mitigation strategy.

The CNS Class 1E station battery rooms are located inside the control building and would not be exposed to extreme low temperatures. At the onset of the event, the Class 1E battery rooms would be at their normal operating temperature and the temperature of the electrolyte in the cells would build up due to the heat generated by the batteries discharging and during recharging. Temperatures in the battery and switchgear rooms are not expected to be sensitive to extreme cold conditions due to their location in the control building, the concrete walls isolating the rooms from the outdoors, and lack of forced outdoor air ventilation during early phases of the ELAP event. However, if operators identified temperatures approaching the design limits of the Class 1E batteries, the licensee would utilize FSG 5.10FLEX.24, "Control Building Temporary Heating FLEX Operations," Revision 0, to establish temporary heating in the control building near the Class 1E battery rooms.

Based on the information above, the NRC staff concludes that the station equipment required to support the FLEX mitigation strategy should perform the required functions at the expected temperatures as a result of loss of heating during an ELAP event consistent with NEI 12-06 Sections 3.2.2.12 and 8.3.2.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable during Phases 2 and 3 is the potential buildup of hydrogen in the Class 1E battery rooms as a result of loss of ventilation during an

ELAP event. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. Once battery charging has commenced, FSGs 5.10FLEX.01, 5.10FLEX.02, 5.10FLEX.03, and 5.10FLEX.04 directs operators to establish portable ventilation for the 903 ft. elevation of the control building.

During Phase 2, operators would deploy three portable ventilation fans that would be powered by a 175 kW FLEX DG to ventilate the battery and switchgear rooms on the 903 ft elevation of the control building. The exhaust path is the same path that is used for the licensee's current station blackout strategy. The portable fans' flow rates are greater than the plant's normal ventilation capability.

Based on its review of the licensee's battery room ventilation strategy, the NRC staff concludes that the hydrogen accumulation in the CNS Class 1E battery rooms should not reach the combustibility limit for hydrogen (4 percent) during an ELAP event.

3.9.2 Personnel Habitability

In the FIP, Section 2.12 describes two areas involving personnel habitability. It describes the control room for which habitability is continuously maintained, and it describes the RCIC room for which personnel entry may be required, but continuous habitability would not be needed. In addition, FIP Section 2.11.1.2 includes operating conditions in the SFP area. These locations are described in the sections that follow.

3.9.2.1 Main Control Room

In the FIP, Section 2.12.1 specifies a personnel temperature limit of 110 °F, and NEDC 15-002, "Portable Equipment Calculations in support of CNS FLEX Strategy," Revision 1, associates this limit with a light work load for an individual. Calculation NEDC 15-002, Revision 1, provides several temperature profiles in the control room over a 72-hour period during an ELAP. These temperature profiles include cases with no compensatory measures and cases with compensatory measures involving opening doors and using fans. The licensee determined that opening doors alone will not prevent room temperature from reaching the limit within 72 hours. Portable ventilation is required. By opening doors at 8 hours and using a portable fan, NEDC 15-002, Revision 1 shows temperatures reaching about 110 °F in 40 hours and reaching about 112 °F after 72 hours. This is just above the limit; however, there is conservatism in these calculations because outdoor air temperature is assumed to be 97°F with no nighttime relief. In addition, the licensee would limit stay times through personnel rotation and implement the controls in Procedure 0.36.1, "Heat Stress Prevention Program." Also, the licensee noted that flexible ducting can be installed on the portable fan to bring in air from the outside and further mitigate temperatures.

3.9.2.2 Spent Fuel Pool Area

In the FIP, Section 2.4.1 states that no Phase 1 actions are required to maintain SFP cooling; however, Phase 2 SFP cooling equipment is staged beginning at about 24 hours into the event (see FIP Section 2.4.2). Establishing a ventilation pathway allows steam and condensate to escape. In order to setup this pathway, the reactor building roof hatch is opened, and as described in FIP Section 2.11.1.2, doors are opened to allow airflow through the SFP area. The roof hatch is opened 3 hours after the ELAP occurs, and as shown in FIP Section 2.17, this is the point when alternate ventilation is established in the reactor building. Ventilation is established prior to SFP boiling which is estimated to occur at 3.82 hours for the full core offload case and at 13.29 hours for the power operation case (see FIP Section 2.4.6).

3.9.2.3 Other Plant Areas

RCIC Room

The RCIC is used continuously in the Phase 1 and 2 reactor core cooling strategies, and although the RCIC room would not need to be continuously occupied, entry may be required. NEDC 15-002, Revision 1 shows the temperature profiles in the room with and without compensatory measures. These profiles show that a long-term temperature of 113°F is achievable with forced ventilation of 5,000 cfm; however, in many of the scenarios with smaller fans, temperatures exceed 120°F. As described in FIP Section 2.12.2, using a small fan and opening doors is sufficient for meeting the equipment's temperature requirement. However, for personnel, protective equipment is needed for entry. In the FIP, Section 2.12.2 states that protective measures such as ice vests will be used if personnel entry is required.

3.9.3 Conclusions

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

3.10 Water Sources

The ECSTs (each nominally 50,000 gallon capacity) contain 97,744 gallons and are protected against all hazards. The suppression pool contains 655,667 gallons and is also protected against all hazards. The hotwell contains 80,625 gallons and also is protected against all hazards. The North well and the Missouri River are unlimited sources of water. The CSTs contain 684,000 gallons combined but are not protected against seismic, flood or high wind hazards. The unqualified water sources may be available during an event, but cannot be credited due to the lack of robustness against one or more of the hazards. Procedures direct the use of high quality water prior to the use of lower quality water. The ECSTs are used first If available and if needed, the hotwell is pumped to the ECSTs via a new permanently installed pump. The North well can then be used to either fill the ECSTs for continued RCIC operation or to provide a source of water for the FLEX pump. The North well water is untreated or "hard" water. While it is free of loose debris that could clog fuel inlet screens, it does contain suspended and dissolved solids, which if used indefinitely, potentially could plate out and foul heat transfer surfaces. In order to minimize this potential fouling, the FLEX Phase 3 strategy will be implemented as soon as NSRC equipment is available to place shutdown cooling in service. This will reduce the well water usage to only to make up for leakage. As a last resort, the FLEX pump can also be supplied from the Missouri River to inject river water to the RPV or SFP.

3.10.1 RPV Make-Up

During Phase 2, RPV makeup supply comes from the ECST initially. When ECST water becomes depleted, the ECST is refilled first from the hotwell using an installed pump, if available, and then from the on-site North well. Alternately, a FLEX pump can be deployed to provide RPV injection of well water through one of two connections. The primary connection is via the normal RHRSW crosstie to the Division 1 RHR injection flow path, and alternately, to the Division 2 RHR torus spray line to the Division 2 RHR injection flow path.

3.10.2 Suppression Pool Make-Up

No makeup to the suppression pool is anticipated to implement the FLEX strategies.

3.10.3 Spent Fuel Pool Make-Up

The primary strategy uses the portable FLEX pump to tie into a FPC system chemical decontamination connection, or alternately, directly to the SFP. The North well pump provides a suction source for the FLEX pump to provide makeup through either of these pathways.

3.10.4 Containment Cooling

In its FIP, the licensee stated that no makeup water is needed for containment temperature or pressure control at any time during the event. Containment temperature and pressure is maintained within limits by venting via the HCVS.

3.10.5 Conclusions

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

3.11 Shutdown and Refueling Analyses

Order EA-12-049 requires that licensees must be capable of implementing the mitigation strategies in all modes. In general, the discussion above focuses on an ELAP occurring during power operations. This is appropriate, as plants typically operate at power for 90 percent or more of the year. When the ELAP occurs with the plant at power, the mitigation strategy initially focuses on the use of the steam-driven RCIC pump to provide the water initially needed for decay heat removal. If the plant has been shut down and all or most of the fuel has been removed from the RPV and placed in the SFP, there may be a shorter timeline to implement the makeup of water to the SFP. However, this is balanced by the fact that if immediate cooling is not required for the fuel in the reactor vessel, the operators can concentrate on providing makeup to the SFP. The licensee's analysis shows that following a full core offload to the SFP, about 44.8 hours are available to implement makeup before boil-off results in the water level in the SFP dropping far enough to uncover fuel assemblies, and the licensee has stated that they have the ability to implement makeup to the SFP within that time.

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

FIP, the licensee stated that it would follow this guidance. During the audit process, the NRC staff observed that the licensee had made progress in implementing this guidance.

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

3.12 Procedures and Training

3.12.1 Procedures

In its FIP, the licensee stated that the inability to predict actual plant conditions that require the use of FLEX equipment makes it impossible to provide specific procedural guidance. As such, the FSGs will provide guidance that can be employed for a variety of conditions. Clear criteria for entry into FSGs will ensure that FLEX strategies are used only as directed for BDBEE conditions, and are not used inappropriately in lieu of existing procedures. When FLEX equipment is needed to supplement EOPs or Abnormal Procedures (APs) strategies, the EOP, AP, Severe Accident Mitigation Guidelines (SAMGs), or Extreme Damage Mitigation Guidelines (EDMGs) will direct the entry into the appropriate FSG procedure. The FSGs will provide available, pre-planned FLEX strategies for accomplishing specific tasks in the EOPs or APs. FSGs are used to supplement (not replace) the existing procedure structure that establishes command and control for the event. Procedural Interfaces have been incorporated into procedural guidance for station blackout to the extent necessary to include appropriate reference to FSGs and provide command and control for the ELAP.

In accordance with site administrative procedures, NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, and NEI 97-04, "Design Bases Program Guidelines", Revision 1, are to be used to evaluate changes to current procedures, including the FSGs, to determine the need for prior NRC approval. However, per the guidance and examples provided in NEI 96-07, Revision 1, changes to procedures (EOPs, APs, EDMGs, SAMGs, or FSGs) that perform actions in response to events that exceed a site's design-basis should screen out. Therefore, procedure steps which recognize the ELAP has occurred and which direct actions to ensure core cooling, SFP cooling, or containment integrity should not require prior NRC approval. The FSGs are reviewed and validated by the involved groups to the extent necessary to ensure that the strategy is feasible. Validation may be accomplished via walk-throughs or drills of the guidelines.

3.12.2 Training

In its FIP, the licensee stated that the nuclear training program has been revised to assure 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 (SAT) Process. Initial training has been provided and periodic training will be provided to site emergency response leaders on BDB emergency response strategies and implementing guidelines. Personnel assigned to direct the execution of mitigation strategies for BDBEEs have received the necessary training to ensure familiarity with the associated tasks, considering available job aids, instructions, and mitigating strategy time constraints. Care has been taken to not give undue weight (in comparison with other training requirements) for operator training for BDBEE accident mitigation. The testing/evaluation of operator knowledge and skills in this area has been similarly weighted. American National Standards Institute/ American Nuclear Society 3.5, "Nuclear Power Plant Simulators for Use in Operator Training," certification of simulator fidelity is considered to be sufficient for the initial stages of the BDBEE scenario until the current capability of the simulator model is exceeded. Full scope simulator models will not be upgraded to accommodate FLEX training or drills.

3.12.3 Conclusions

Based on the description above, the NRC staff concludes that the licensee has adequately addressed procedures and training associated with FLEX. The procedures have been issued in accordance with NEI 12-06, Section 11.4, and a training program has been established and will be maintained in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

In its FIP, the licensee stated that the maintenance and testing of FLEX equipment is governed by the CNS preventative maintenance (PM) program and is consistent with INPO AP-913. This program utilizes the EPRI preventive maintenance basis database as an input in development of the CNS PM basis templates. Based on this, the CNS PM program for FLEX equipment follows the guidance in NEI 12-06, Section 11.5.

The CNS PM basis templates include activities such as:

- Periodic static inspections
- Operational inspections
- Fluid analysis
- Periodic functional verifications
- Periodic performance verification tests

The CNS PM basis templates provide assurance that stored or pre-staged FLEX equipment is being properly maintained and tested. In those cases where EPRI templates were not available for the specific component types, PM actions were developed based on manufacturer provided information/ recommendations. Additionally, the emergency response organization (ERO) performs periodic facility readiness checks for equipment that is outside the jurisdiction of the normal PM program and considered a functional aspect of the specific facility (EP communications equipment such as power supplies, radios, batteries, battery chargers, satellite phones, etc.). These facility functional readiness checks provide assurance that the EP communications equipment outside the jurisdiction of the PM program is being properly maintained and tested. The unavailability of equipment and applicable connections that directly perform a FLEX mitigation strategy for core, containment, and SFP is managed such that risk to mitigating strategy capability is minimized. Maintenance/risk guidance conforms to the guidance of NEI 12-06 as follows:

- Portable FLEX equipment may be unavailable for 90 days provided that the site FLEX capability (N) is available.

- If portable equipment becomes unavailable such that the site FLEX capability (N) is not maintained, initiate actions within 24 hours to restore the site FLEX capability (N) and implement compensatory measures (e.g., repair equipment, use of alternate suitable equipment or supplemental personnel) within 72 hours. Work Management procedures will reflect allowed outage times as outlined above.

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 (ADAMS Accession No. ML13276A573), which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 (ADAMS Accession No. ML13276A224), the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs. As stated above, the licensee utilized the EPRI preventive maintenance basis database as an input in development of the CNS PM basis templates.

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

3.14 Conclusions for Order EA-12-049

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

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 (ADAMS Accession No. ML13070A007), the licensee submitted its OIP for CNS in response to Order EA-12-051. By letter dated September 12, 2013 (ADAMS Accession No. ML13256A082), the NRC staff sent a request for additional information (RAI) to the licensee. The licensee provided a response by letter dated October 14, 2013 (ADAMS Accession No. ML13294A027). By letter dated December 4, 2013 (ADAMS Accession No. ML13323A105), the NRC staff issued an Interim Staff Evaluation (ISE) and RAI to the licensee.

By letters dated August 27, 2013 (ADAMS Accession No. ML13247A281), February 26, 2014 (ADAMS Accession No. ML14064A265), August 26, 2014 (ADAMS Accession No. ML14246A204), February 23, 2015 (ADAMS Accession No. ML15062A038), August 27, 2015 (ADAMS Accession No. ML15253A370), February 16, 2016 (ADAMS Accession No. ML16054A798), and August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee submitted status reports for the Integrated Plan and the RAI in the ISE. The Integrated Plan describes the strategies and guidance to be implemented by the licensee for the installation of reliable Spent Fuel Pool Level Instrumentation (SFPLI) which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated December 20, 2016 (ADAMS Accession No. ML17234A314), the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed a SFPLI system designed by Mohr Test and Measurement LLC. The NRC staff reviewed the vendor's SFPLI system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report on August 27, 2014 (ADAMS Accession No. ML14216A362).

The staff performed the onsite audit to review the implementation of SFPLI related to Order EA-12-051. The scope of the audit includes verification of (a) site's seismic and environmental conditions enveloped by the equipment qualifications, (b) equipment installation met the requirements and vendor's recommendations, and (c) program features met the requirements. By letter dated August 29, 2016 (ADAMS Accession No. ML16217A475), the NRC issued the audit report on the licensee's progress. Refer to Section 2.2 above for the regulatory background for this section.

4.1 Levels of Required Monitoring

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].

In its OIP, the licensee identified the SFP levels of monitoring as follows:

- Level 1 corresponds to the 999 feet (ft.) 7 inches (in.) plant elevation
- Level 2 corresponds to the 987 ft. 5-3/8 in. elevation
- Level 3 corresponds to the 977 ft. 2-3/8 in. elevation

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee provided a sketch depicting the SFP levels of monitoring as illustrated below in Figure 1, "Cooper Nuclear Station SFP Levels of Monitoring."

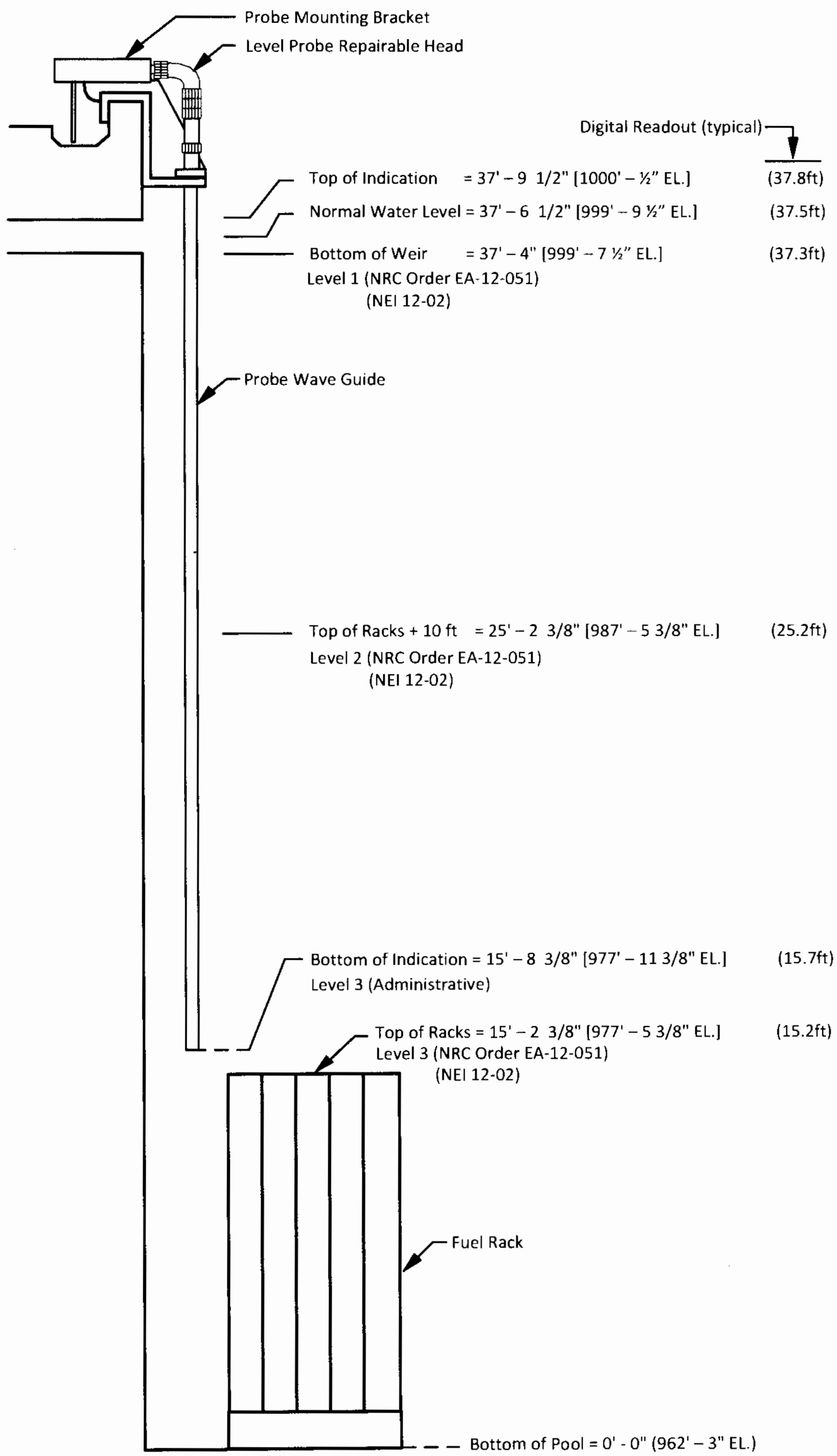

Figure 1 – Cooper Nuclear Station SFP Levels of Monitoring

The NRC staff's assessment of the licensee's selection of the SFP levels of monitoring is as follows.

- Level 1: Level 1 at 999 ft. 7 in. plant elevation is adequate for normal SFP cooling system operation and it is also adequate to ensure the required fuel pool cooling pump net positive suction head (NPSH) as the skimmer surge tanks supply the SFP cooling pumps. This level represents the higher of the two points described in NEI 12-02 for Level 1.

- Level 2: Level 2 was identified by the licensee as 987 ft. 5-3/8 in. plant elevation. This level is consistent with the first of the two options described in NEI 12-02 for Level 2, which is 10 feet (+/- 1 foot) above the highest point of any fuel rack seated in the SFP.

- Level 3: Level 3 was identified by the licensee as 977 ft. 5-3/8 in. plant elevation, which is the highest point of any fuel rack seated in the SFP where fuel remains covered; and therefore, consistent with NEI 12-02.

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

4.2 Evaluation of Design Features

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

4.2.1 Design Features: Instruments

In its OIP, the licensee stated that the primary and back-up instrument channels will be of identical design and will consist of fixed components. Both channels utilize guided wave radar (GWR), which functions according to the principle of time domain reflectometry (TDR). This technology requires that a wave guide for each channel be installed in the pool water such that it runs the entire length of the instrument range.

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee provided a sketch (Figure 1 - Cooper Nuclear Station SFP Levels of Monitoring) depicting the SFP level instrument's measuring range from 1000 ft. ½ in. to 977 ft. 11 3/8 in. plant elevations. The NRC staff noted that the instrument's measuring range (1000 ft. ½ in. elevation – 977 ft. 11 3/8 in. elevation) covers Levels 1 and 2 but does not cover Level 3 (977 ft. 5-3/8 in. elevation). In response to the staff's concern, in an email dated March 1, 2017 (ADAMS Accession No. ML17075A105), the licensee noted that Administrative Level 3 (977 ft. 11-3/8 in. elevation) is used in CNS's operation and calibration procedures. According to Procedure EOP 5A, "Secondary Containment Control," Revision 18, if the SFP water level drops 4 in. below the normal water level, operators will take action to restore and maintain SFP water level between 25 ft. 3 in. (987 ft. 6 in. elevation) and 37 ft. 9 in. (1000 ft. 0 in.elevation). If the water level continues to drop, EOP 5A directs operators to take actions to initiate SFP spray before the water level drops to 15 ft. 8.4 in. (Level 3 administrative - 977 ft. 11-3/8 in. elevation). This procedure reading is within the span of the SFP level instrument's measuring range. Since initiating water make-up prior to SFP water level reaching Level 3, the staff concludes that the actions to initiate water make-up are not delayed unnecessarily, and therefore, consistent with NEI-12-02 guidance.

The NRC staff concludes that the licensee's design, with respect to the number of SFP instrument channels and instrument's measuring ranges, appears to be consistent with the intent of NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.2 Design Features: Arrangement

Regarding CNS SFPLI level probe arrangement, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that NEI 12-02, Section 3.2, discusses the location requirements of the sensors which include opposite sides or corners of the pool area or separated by a distance comparable to the shortest length of a side of the pool. The location of the sensors for CNS' pool meets or exceeds these requirements as they are located as close as practical to the opposite comers and are separated at a distance comparable to the longer sides of the pool (40 ft.). This configuration provides the maximum separation for the physical location of the sensors in the pool and meets the criteria specified in NEI 12-02, Section 3.2. In the same letter, the licensee provided a sketch depicting the location of the SFPLI level probe. The primary channel probe is located near the northeast corner of the pool, and the back-up channel probe is located near the southwest corner of the pool.

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the electronics package (i.e., signal processor) for each channel of the SFPI system is located separately from that of the other channel. The signal processor for the primary channel is located in the cable spreading room which is within the control room emergency filter system boundary. The signal processor for the back-up channel is located in the control building corridor on the ground floor. This separation will prevent failure of both channels due to a common missile or other physical threat.

In its OIP, the licensee stated that the power and signal cable required for each channel will be routed separately from the other channel. Conduit supports that are qualified for seismic Class I applications will be used for routing all conduit in both the reactor and the control buildings. The conduit in the reactor building will be installed to ensure that it will not interfere with fuel handling activities or other activities in the SFP.

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

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

4.2.3 Design Features: Mounting

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the SFP level indication probe demonstrates compatibility with CNS' seismic design-basis. The qualification of the probe components and mounting bracket meets the qualification standards outlined in Institute of Electrical and Electronics Engineers (IEEE) Standard 344-2004 and Section III of ASME Boiler and Pressure Vessel Code (2008). A combination of hydrodynamic and static analyses was performed (NEDC 14-017, "Seismic Induced Hydraulic Response in the CNS SFP") to determine the estimated total loading on the probe and the mounting device to support the SFP probe. Generic hydrodynamic analysis of the controlling cases for the utility services alliance, and site specific analysis including the dimensions of the CNS SFP, was performed using the computer code GOTHIC 8.0. The site specific analysis was performed by exciting the SFP pool inventory with acceleration time-histories in each direction. The vertical 3D fluid velocity of the SFP fluid was extracted from this analysis and applied to the mounting bracket for stress analysis of the probe support system. A finite element fluid-structure interaction (FSI) analysis of the probe and support system was performed (NEDC 14-018, "MOHR SFP-1 Site-Specific Seismic Analysis Report for CNS") in addition to the GOTHIC analysis. The 3-D time-history FSI analysis was performed using ANSIS Mechanical software. The results from this analysis were used to perform the stress qualification of the various components of the probe. The support reactions from the FSI analysis were used as input in the stress qualification of the mounting bracket which was also performed using the ANSIS Mechanical software.

Related to the mounting design for other SFPLI equipment including the electronics enclosures, battery enclosures, and power conditioners, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that both the primary and back-up channel electronics enclosures, battery enclosures, power conditioners, and uncredited Yokogawa indicators were seismically mounted per requirements of NEDC 14-005, "Conduit Routing and Support Design."

As for other mounting design for the SFPLI conduit support, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the hanger supports for the conduit installation were seismically mounted in accordance with:

- NEDC 14-005, "Conduit Routing and Support Design"
- NEDC 11-119, "Qualification of Existing Conduit Support 07-15 (Drawing EERBH3007-EE-RBH3015) for ADHR Instrumentation Conduit"
- NEDC 87-080, "Seismic Design Calculation for Conduit Hanger and Electrical Junction Box Supports for Torus Level Indicator"
- NEDC 15-068, "Conduit Hangers CBH2042, CBH2041, and CBH2069 Seismic Analysis"
- NEDC 90-065, "Seismic Analysis for RBH 2060, RBH 2063, RBH 1275, RBH 1276, RBH 1806T, and New STD Type F"
- NEDC 87-152, "Hanger Calculations for CNS Battery Replacement Project"

The licensee further stated that the control building and the reactor building are the structures that the primary and secondary (back-up) instrument channels were routed and secured to. Both buildings are seismic Clas I structures. As such, all equipment is mounted to seismic Class I criteria.

The NRC staff noted that the licensee adequately addressed the design criteria and methodology used to estimate and test the total loading on the mounting devices, including the design-basis maximum seismic loads and the hydrodynamic loads that could result from pool sloshing. The site-specific seismic analyses demonstrated that the SFP level instrumentation's mounting design is satisfactory to allow the instrument to function per design following the maximum seismic ground motion. The assumptions, analytical, and model used in the sloshing analysis for the sensor mounting bracket are adequate. The staff also noted that the licensee adequately addressed the design inputs and methodology used to qualify the structural integrity of the affected plant structures.

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

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

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

In its OIP, the licensee stated that the reliability of the instrumentation would be established through the use of an augmented quality assurance process similar to that applied to the site fire protection program.

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

4.2.4.2 Equipment Reliability

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

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

- seismic effects on instrument channel components used during and following a potential seismic event for only installed components.

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

During the vendor audit, the NRC staff reviewed the Mohr SFPLI's qualifications and testing for temperature, humidity, radiation, shock and vibration, seismic, and electromagnetic compatibility. The staff further reviewed CNS's seismic, radiological, and environmental anticipated conditions to verify the CNS bounding conditions. Below is the staff's assessment of the equipment reliability of the CNS SFPLI.

4.2.4.2.1 Radiation, Temperature, and Humidity

Spent Fuel Pool Area

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee provided the equipment design limits and the BDB radiological and environmental conditions of the CNS reactor building as shown below in Table 1, "CNS SFP Area Environmental and Radiological Conditions vs. Equipment Design Limits."

Table 1 – CNS SFP Area Radiological and Environmental Conditions vs. Equipment Design Limits

Parameters	CNS BDB Conditions	Equipment Limits
Radiation	Total Integrated Dose 1.15 x 10^9 rad	2.0 x 10^9 rad
Temperature	212°F (pool) 188°F (general area)	212°F
Humidity	100%	100%

The licensee further stated that Calculation NEDC 13-030, "Spent Fuel Pool Instrumentation Total Integrated Dose Calculation," quantified the total integrated dose (TID) for the EPDM [ethylene propylene diene monomer] rubber insulator located inside the wall of the MOHR liquid level measurement probe. The probe is mounted in the SFP, and the TID over 30 years and 7 days of Level 3 accident scenario (pool water even with the top of the spent fuel racks) is calculated to be 1.15 x 10^9 rad at 3 ft. above the rack. This level is significant in that it is the closest installation of the EPDM rubber spacer within the probe. The EPDM rubber is qualified for a radiation dose of 2.0 x 10^9 rad. Therefore, the dose of 1.15 x 10^9 rad is encompassed.

Outside of SFP Area

For radiological and environmental conditions of the cable spreading room and the corridor of the control building, where the primary and back-up SFPLI electronics equipment are located, respectively, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the signal processor and battery packs are not located in the vicinity of the SFP. Therefore, there is no need to qualify them for the temperature, humidity, water chemistry, or radiation produced by boiling the SFP. According to the licensee, the equipment design limits envelop the CNS radiological and environmental conditions of the control building as shown below in Table 2, "CNS Control Building Environmental Conditions vs. Equipment Design Limits."

Table 2 – CNS Control Building Environmental Conditions vs. Equipment Design Limits

	Parameters	CNS BDB Conditions	Equipment Limits
Cable Spreading Room (918) (Primary Channel)	Temperature	120°F	123°F
	Humidity	60%	95%
903 Corridor (Back-up Channel)	Temperature	117.5°F	123°F
	Humidity	60%	95%

The licensee further stated that the heat-up design-basis calculation is NEDC 93-054, "Control Room Heatup During 24 Hour Period," and the calculation for the BDB heat-up is NEDC 15-002, "Review of Tetra Tech Portable Equipment Calculations in Support of CNS FLEX Strategy." NEDC 88-299A, "Review of S&L Calc. No. COOLC-01, Revision 6, HVAC Load Calculation for Control Building EL 903'-6"," Revision 9, calculated the temperature for the control building 903 corridor at 117.5°F during a loss of non-essential ventilation. The analysis of the 903 control building corridor heat-up for ELAP (secondary channel) is bounded by this 117.5°F for the ELAP and with a 60 percent relative humidity (RH) (Tetra Tech Calculation 194-4959-02 contained in NEDC 15-002). The temperature in the analysis is 101.4°F, which is very near its steady state temperature. The analysis of the 918 control building cable spreading room heat-up for ELAP (primary channel) is bounded by 120°F for the ELAP and with a 60 percent RH (Tetra Tech Calculation 194-4959-02 contained in NEDC 15-002). The temperature in the analysis is 99.56°F in a 24-hour time frame, which is very near its steady-state temperature.

The staff concludes that the licensee adequately addressed the equipment reliability of SFP level instrumentation with respect to temperature, humidity and radiation. The equipment design limits envelop the CNS's anticipated conditions of radiation, temperature, and humidity during a postulated BDBEE and post event. The equipment environmental testing demonstrated that the SFP instrumentation should maintain its functionality under expected BDB conditions.

4.2.4.2.2 Shock and Vibration

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the SFPLI system was tested by the manufacturer, MOHR Test & Measurement, LLC. NEDC 14-009 accepts MOHR Document 1-0410-5, "MOHR EFP-IL SFPI System Shock and Vibration Test Report," which reports the shock and vibration test for the signal processor and back-up battery pack based on IEC 60068-2-27 (for shock requirements) and IEC 60068-2-6 (for vibration requirements).

The NRC staff noted that the licensee adequately addressed the equipment reliability of SFPLI with respect to shock and vibration. The NRC performed an audit at Mohr facility for shock and vibration testing of the SFPLI sensors and electronics components. The staff found them acceptable. The staff concludes that the licensee's proposed design with respect to shock and vibration, if implemented appropriately, should provide its design functions.

4.2.4.2.3 Seismic

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated, in part, that in order to provide reasonable assurance that the instrumentation will provide reliable service in a nuclear power plant, the system must demonstrate compatibility with anticipated seismic effects according to relevant standards. The IEEE Standard 344-2004 was used and was referenced to develop a suitable seismic test plant meeting the requirements of IEEE 344-2004 and triaxial shake-table testing to 10 CFR Part 50, Appendix B quality requirements. The seismic testing is documented in the EFP-IL SFPI System Seismic Test Report, MOHR Document 1-0410-6 (NEDC 14-010). This testing included taking baseline functional data, visual observation of the equipment during seismic testing, and collection of post-seismic functional data. This testing was performed at both the site-specific test levels as well as the limits of the test table.

The NRC staff noted that the licensee adequately addressed the equipment reliability of SFPLI with respect to seismic. The SFPLI was tested to the seismic conditions that envelop CNS' design-basis maximum ground motion. Further seismic qualifications of the SFPLI mounting is addressed in Subsection 4.2.3, "Design Features: Mounting," of this evaluation.

The NRC staff concludes that the CNS SFPLI qualification process to be adequate. However, the staff has learned that there were incidents at other nuclear facilities, in which the MOHR's SFPLI experienced failures of the filter coil (or choke). The staff requested CNS to address the impact of MOHR SFPLI failures on its equipment. The licensee provided a response in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), in which it stated that Qualification Report "EFP-IL MOD 1 Modification Package," provides the evaluations of the replacement parts. Section 3.2 of this document specifically addresses the choke failure and the solution for the fix. Cooper's SFPI equipment was upgraded and tested prior to the system installation.

The NRC staff's assessment of the Modification Package MOHR EFP-IL MOD 1, Revision 0, dated July 16, 2015. is summarized as below.

In EFP-IL MOD 1, MOHR provided evaluation of the following hardware modifications:

- In-place replacement of the miniature T1 surface mount choke on the 01-EFP-IL-50001 board with an equivalent component.
- In-place replacement of the miniature T1 surface mount choke on the 01-EFP-IL-50006 board with an equivalent component.
- Incorporation of a fusible link in the 01-EFP-IL-50204 cable assembly.
- Full electrical isolation added to the 01-EFP-IL-50007 (USB interface) board.

Below is the summary of the vendor's evaluation of the above modifications:

T1 Choke Replacement Evaluation

Temperature and Humidity
The replacement choke has an operating temperature range of -40 °C to +85 °C, exceeding the -10 °C to +55 °C requirement. Non-condensing humidity does not alter performance of this component.

Electromagnetic Compatibility (EMC)
There is no change to EMC qualification. The choke demonstrates equivalent or higher impedance to common mode noise.

Shock and Vibration
The mass differences are 0.002% and 0.47% for 01-EFP-IL-50001 enclosure mass and the board mass respectively and 0.0003% and 0.18% for 01-EFP-IL-50006 enclosure mass and the board mass respectively.

Seismic
Qualification by similarity to existing qualified equipment is permitted by IEEE Std. 344-2004. Replacement of T1 choke does not significantly alter equipment mass, mass distribution, or other mechanical characteristics.

Fusible Link Evaluation:
The 01-EFP-IL-50204 Fusible Link is added to the existing power board power cable. One Fusible Link is used per EFP-IL signal processor.

Temperature and Humidity
The Fusible Link's fuses are rated for -55 °C to +125 °C and 100% relative humidity per MIL-STD-201 Method 106.

Electromagnetic Compatibility (EMC)
There is no change to EMC qualification. The Fusible Link uses insulated wiring and connectors identical in configuration to the remainder of the previously qualified cable assembly and expected emissions are unchanged.

Shock and Vibration
The Fusible Link uses insulated wiring and connectors identical in configuration to the remainder of the cable assembly and is secured using identical tie-down and strain-relief methods which have been previously qualified.

The Fusible Link contributes approximately 0.14% enclosure mass, well within the expected variation of the unmodified EFP-IL signal processor enclosure mass due to manufacturing tolerances of system components.

The connectors are qualified by the manufacturer for vibration conditions per EIA 364-28 and shock loading at 50g. The connector rated minimum pull force is 8.0 lbf per wire terminal, for a total rating of 80.0 lbf for the 10 wire connector, equivalent to 2086g loading assuming Fusible Link mass of 17.4 g.

The Littelfuse fuse lead axial pull force is rated at 7 lbs. per MIL-STD-202, which is equivalent to 182 g static loading per fuse (two fuses per cable), assuming cable mass of 17.4 g and conservatively neglecting stress shielding by cable wiring and insulation.

Seismic
The Fusible Link contributes approximately 0.14% enclosure mass, well within the expected variation of the unmodified EFP-IL signal processor enclosure mass due to manufacturing tolerances of system components.

Electrical Isolation Evaluation:
The front panel USB board 01-EFP-IL-50007 has been modified through addition of the component to provide galvanic USB isolation and enhance ESD protection (±15kV).

Temperature and Humidity
The additional component is rated for normal operation at -40 °C to +85 °C. The component is a hermetic plastic BGA package that is not susceptible to elevated humidity.

Electromagnetic Compatibility (EMC)
There is no change to EMC qualification. The isolation technology within the additional component is compliant with applicable standards including radiated emissions limit per IEC 61000/CISPR 22. The device reduces the equipment's already low radiated emissions when the USB device is in use because it isolates and prevents noise on internal data and power lines from propagating to external devices connected to the front-panel USB port. The device is not active when USB devices are not in use.

Shock and Vibration
There is insufficient mass difference to alter the equipment shock and vibration response characteristics. The additional component is a rugged, compact, encapsulated surface-mount BGA package enveloped in size and mass by other components in the system. Surface mount components as a class are not susceptible to required levels of shock and vibration when mounted within the EFP-IL equipment enclosures.

Seismic
The nominal difference in enclosure mass is trivial at 0.014, well within the expected variation of EFP-IL signal processor enclosure mass due to manufacturing tolerances of system components.

The NRC staff found that the vendor adequately addressed the staff concerns with regard to the modified equipment qualifications. The temperature and humidity ratings of the replacement parts envelop the expected BDB environmental conditions of CNS' control building areas where the electronics equipment is located. Based on the vendor report, there is no indication that new electromagnetic emissions are introduced by the replacement parts. The mass differences are insufficient to alter the seismic, shock, and vibration response characteristics.

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

4.2.5 Design Features: Independence

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the level sensors and fixed components of each channel on the refueling floor are physically separated from the other channel by a distance that is comparable to the [longer] side of the SFP. The signal processor, uninterruptible power supply (UPS), and back-up battery packs for each channel are located in diverse locations. The signal processor for the primary channel is located in the cable spreading room which is within the control room emergency filter system boundary. The signal processor for the back-up channel is located in the control room. For areas other than the refueling floor, no minimum physical separation distance is applied beyond that which is normally required to maintain the separation of the two divisions. Cable for each channel is routed in conduit and cable trays that are separated from the other channel. The Division 1 tray is used for the primary channel, and the Division 2 tray is used for the back-up channel.

Regarding the channel's electrical independence, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that each channel is powered from a separate and independent source from the other. The primary channel is powered from the UPS for the plant management information system (PMIS). The back-up channel is powered from a Division 2 lighting panel in the auxiliary relay room.

The NRC staff noted, and verified during the walkdown, that the licensee adequately addressed the SFPLI channel independent. The instrument channels' physical separation is further discussed in Subsection 4.2.2, "Design Features: Arrangement." With the licensee's proposed design, the loss of one level instrument channel would not affect the operation of the other channel under BDBEE conditions. The staff concludes that the licensee's proposed design, with respect to instrument channel independence, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.6 Design Features: Power Supplies

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the electrical ac power source for the secondary channel is powered from lighting panel LPCB2. The power path for LPCB2 is via 4160V critical bus 1F to (Division 1) 480V critical bus 1F to MCC F to LPCB2. The electrical ac power source for the primary channel is powered from PMIS UPS main panel UPS1A. The power path for EE-PNL-UPS1A is via EE-XFMR-MPF2 to EE-SWBR-MPF2, 480V MDP2 to EE-PNL-UPS1A. This supply also has an emergency feed to 480V MCC L. Other power sources used in the two channels are dc via the Mohr UPS units or via additional battery external connections.

As for the instrument battery's duty cycle, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that Mohr Test & Measurement LLC has tested the battery packs with the EFP-IL system to show that in the "Low Power" mode of operation, the batteries will supply 7 days of power provided that the SFP level measurements are restricted to no more than 15 samples per hour. This sample rate is programmed into the "Low Power" configuration of the system to ensure this rate is maintained. The test results are documented in MOHR Document 1-0410-7 (NEDC 14-011, "MOHR EFP-IL SFPI Battery Life Report").

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

4.2.7 Design Features: Accuracy

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the acceptance criteria for accuracy of the level indication system provided in the vendor's testing documentation is ± 1 inches. This is not affected by the water level, so this accuracy holds throughout the range of the instrument. The accuracy that will be required of this system by CNS is ± 3 inches. Under BDB conditions, the accuracy shift that occurs can be expected to be within the maximum deviations that were observed during qualification testing by the vendor. The maximum deviations that were observed during testing were:

- Seismic: +0.1 in. (MOHR Document 1-0410-6)
- Temperature: -0.4 in. (MOHR Document 1-0410-1)
- Humidity: -0.2 in. (MOHR Document 1-0410-1)
- Shock/Vibration: +0.2 in. (MOHR Document 1-0410-5)

Only the probe may be exposed to significant amounts of radiation. Since it acts only as a wave guide, it will have no impact on the accuracy of the system so long as it remains intact. Thus, radiation will not affect the accuracy of the system. The calibration tolerance being specified for the CNS system is ± 3 inches.

The NRC staff noted that the licensee adequately addressed the SFPLI accuracy design including the expected instrument channel accuracy performance under both normal and BDB conditions. If implemented properly, the instrument channels should maintain the designed accuracy following a change or interruption of power source without the need of recalibration. The staff concludes that the licensee's proposed instrument accuracy appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately addresses the requirements of the order.

4.2.8 Design Features: Testing

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that CNS uses Mohr equipment which provides the capability to perform in-situ testing. CNS uses Procedures 14.41.1.1, "FPC-LIT-1 Testing," and 14.41.1.2, "FPC-LIT-2 Testing," to perform testing and calibration associated with the Mohr equipment. Periodic testing and calibration is performed at the signal processor using standard calibration equipment (time-domain reflectometer, oscilloscope, 500 male TNC terminator, test cable).

Regarding the instrument channel check, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that channel checks are enabled by regular readings taken by operations personnel from each channel and actual fuel pool level obtained locally. A weekly channel check by operations is performed, comparing the level reading from each channel with actual fuel pool level. These checks are incorporated into the plant surveillance program. The channel checks are performed per Procedures 2.1.12, "Control Room Data," 2.1.11.1, "Turbine Building Data," and 2.1.11.2, "Reactor Building Data."

For the SFPLI functional test, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that a channel functional test will be performed prior to each planned refueling outage. The SFPLI is functionally tested using both onboard generated test signals and calibrated test equipment.
For the instrument calibration, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that a calibration test will be performed at the frequency recommended by the vendor manual. The calibration check will use the calibration menu to enable inserting a 4-20mA signal to the recorder and the indicator. In addition, the calibration will incorporate a two point check from the probe using the shorting pin and actual level as the two points.

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

4.2.9 Design Features: Display

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the main indication for the primary channel is on the signal processor located on the south side of the cable spreading room below the control room. This channel feeds a secondary signal to a remote indication on an existing control room recorder, RHR-TR-131. The remote indication located in the control room is not credited to meet the order. The credited indication is on the Mohr equipment. The main indication for the back-up channel is on the signal processor located on the south wall of the control building corridor on the ground floor (903 ft. elevation). This channel feeds a secondary signal to a remote outdoor indicator on the outside wall of the northeast comer of the control building. The remote outdoor indicator is not credited to meet the order. The credited indication is on the Mohr equipment.

Regarding the display locations, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the primary channel has a readout in the control room and one in the cable spreading room, which is located directly below the control room and is easily accessible. The primary channel display will be periodically monitored. The secondary channel has an outdoor remote readout located on the northeast comer of the control building. This is intended for use by personnel that are providing make-up water to the SFP from a location that is remote from the SFP as required by NRC Order EA-12-049 and NEI 12-06. The indication on the secondary signal processor is located on the ground floor of the control building in the corridor, which is easily accessible from the outdoor indication via the turbine building. The operator's path to get to the credited indication is similar. The secondary credited indicator is the further of the two credited indicators (located on the 903 ft. control building vs 918 ft. control building). The conservative time frame for the operator to reach the secondary indicator is under a 15 minute time frame taking account for any potential loss of lighting and obstructions caused by debris.

Regarding habitability of the display locations, in its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the display locations are considered to be mild environment areas and are well within vendor specifications for the equipment. The design-basis heat-up calculation is NEDC 93-054 and for the BDB heat-up is NEDC 15-002. NEDC 88-299A establishes the maximum allowable temperature for the control building 903 ft. corridor at 117.5°F during loss of non-essential ventilation. The analysis of the 903 ft. control building corridor heat-up (secondary channel) is bounded by this 117.5°F for the ELAP and with a 60 percent RH. The temperature in the analysis is 101.4°F, which is very near its steady-state temperature. The analysis of the 918 ft. control building cable spreading room heat-up (primary channel) is bounded by 120°F for the ELAP and with a 60 percent RH. The temperature in the analysis is 99.56°F in a 24-hour time frame, which is very near its steady state temperature. No radiological concerns exist for these areas during an ELAP event.

The NRC staff concludes that the licensee's proposed location and design of the SFPLI displays appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3 Evaluation of Programmatic Controls

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that SAT methods will be used to identify the population to be trained and to determine both the initial and continuing elements of the required training.

Guidance document NEI 12-02 specifies that the SAT process can be used to identify the population to be trained, and also to determine both the initial and continuing elements of the required training. The NRC staff concludes that the licensee's plan to train personnel in the operation, maintenance, calibration, and surveillance of the SFP level instrumentation, including the approach to identify the population to be trained, appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3.2 Programmatic Controls: Procedures

In its OIP, the licensee stated that procedures will be developed or revised, as necessary, using guidelines and vendor instructions to address the maintenance, operation, and abnormal response issues associated with the new SFP level instrumentation. In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee noted that the following SFPLI-related procedures have been developed:

- Procedure 2.1.11.1, "Turbine Building Data" - Equipment operator logs to record readings from the EFP-ILs.
- Procedure 2.1.11.3, "Radwaste and Augmented Radwaste Building Data" – Equipment operator logs to record readings from the back-up channel remote readout.
- Procedure 2.1.12, "Control Room Data" - Control room logs to record readings on the primary channel remote readout.
- Procedure 2.2A-120LTG.DIV0, "Station Lighting System Non-Divisional Power Checklist" - Breaker lineup checklist for power to back-up channel.
- Procedure 2.2.32, "Fuel Pool Cooling and Demineralizer System" - Operation of the fuel pool cooling and demineralizer system, including the EFP-ILs.
- Procedure 2.2.32A – "Fuel Pool Cooling and Demineralizer System Component Checklist" - Component checklist for 2.2.32.

- Procedure 2.4FPC, “Fuel Pool Cooling Trouble” - Abnormal response procedure for fuel pool cooling.

- Procedure 2.2.63A, “PMIS Uninterruptible Power Supply Component Checklist” – Breaker lineup checklist for power to primary channel.

- Procedure 2.3_9-4-2, “Panel 9-4 - Annunciator 9-4-2” - Verifies pool level when fuel pool trouble alarms due to level.

- Procedure 5.1RAD, “Building Radiation Trouble” - Will utilize remote indication to verify level of the pool when high radiation levels occur in the area.

- Procedure 14.41.1, “Fuel Pool Cooling System Instrument Calibration” – Provides instructions to calibrate fuel pool cooling system instruments.

- Procedure 14.41.1.1, “FPC-LIT-1 Testing” - Provides instructions to perform instrument calibrations on the spent fuel pool FLEX level indicating loop and transmitter FPC-LIT-1.

- Procedure 14.41.1.2, “FPC-LIT-2 Testing” - Provides instructions to perform instrument calibrations on the secondary spent fuel pool FLEX level indicating loop and transmitter FPC-LIT-2.

- Procedure 6.LOG.601, “Daily Surveillance Log - Modes 1, 2, and 3” - Provides instructions for operations personnel to perform high frequency technical specifications (TS), technical requirements manual (TRM), and off-site dose assessment manual (ODAM) surveillance requirements during Modes 1, 2, and 3.

- Procedure 6.LOG.602, “Daily Surveillance Log - Modes 4 or 5” - Provides instructions for operations personnel to perform high frequency TS, TRM, and ODAM surveillance requirements during Modes 4 or 5.

- Procedure 5.3SBO, “Station Blackout” - Provides operator guidance for a loss of all ac power (on and offsite). Attachment 2, “RPV and Containment Parameter Monitoring,” has a proposed change to install a back-up battery to the SFPLI per FSG 5.10FLEX.06.

- FLEX Support Guideline 5.10FLEX.06, “Fuel Pool Level Instrument Electrical Tie-In” - Establishes 12 Vdc back-up power supply for continued use of SFP level instrumentation.

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

4.3.3 Programmatic Controls: Testing and Calibration

In its OIP, the licensee stated that processes will be established and maintained consistent with the applicable NEI 12-02 guidelines for scheduling and implementing necessary testing and calibration of the primary and backup SFP level instrument channels to maintain the instrument channels at the design accuracy. The testing and calibration of the instrumentation will be consistent with vendor recommendations or other documented basis.

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee further described the CNS testing and calibration program. Instrument channel checks are enabled by regular readings taken by operations personnel from each channel and actual fuel pool level obtained locally. A weekly channel check by operations is performed, comparing the level reading from each channel with actual fuel pool level. These checks are incorporated into the plant surveillance program. A functional test will be performed prior to each planned refueling outage. This functional test will be incorporated into CNS' surveillance program. A calibration test will be performed at the frequency recommended by the vendor manual. This calibration test will be incorporated into the plant surveillance program.

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the PM plan will consist of maintenance items with 6-month, 12-month, and 24-month maintenance frequency (MF) actions as follows:

- 6-month MF actions include memory test, battery test, scan test
- 12-month MF actions include loop calibration
- 24-month MF actions include battery replacements, internal clock battery verification, drift check, memory card replacement, and probe and cable diagnostics scan

In its letter dated August 22, 2016 (ADAMS Accession No. ML16245A290), the licensee stated that the primary or back-up instrument channel can be out-of-service for testing, maintenance, and/or calibration for up to 90 days provided the other channel is functional. Additionally, compensatory actions must be taken if the instrumentation channel is not expected to be restored or is not restored within 90 days. For a single channel that is not expected to be restored, or is not restored within 90 days, the compensatory actions will include steps necessary to ensure availability of normal alarms and proper function of the remaining indication channel validated by direct visual monitoring. If both channels become non-functional, then actions will be initiated within 24 hours to restore one of the channels of instrumentation and to implement compensatory actions within 72 hours. Compensatory actions will include steps necessary to ensure availability of normal alarms and increased direct visual monitoring of the SFP level.

According to the licensee, once the SFPI is incorporated into the TRM as part of the FLEX strategy, the required actions and completion times will be driven by the TRM (as shown below in Table 3, "CNS TRM T3.12.6 – Spent Fuel Pool Level Instrumentation")

Table 3 – CNS TRM T 3.12.6 – Spent Fuel Pool Level Instrumentation

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The primary or back-up spent fuel pool level instrument does not meet the functional requirements.	A.1 Restore spent fuel pool level instrument to functional status	90 days
B. Action A.1 completion time not met.	B.1 Implement compensatory measures.	Immediately
C. The primary and back-up spent fuel pool level instruments do not meet the functional requirements.	C.1 Initiate actions to restore one of the channels of instrumentation. AND C.2 Implement compensatory measures.	24 hours 72 hours

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

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

4.4 Conclusions for Order EA-12-051

In its letter dated December 20, 2016 (ADAMS Accession No. ML17234A314), the licensee stated that they would meet the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In the evaluation above, the NRC staff concludes that the licensee has conformed to the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. Based on the evaluations above, the NRC staff concludes that if the SFPLI is installed at CNS according to the licensee's proposed design, it will adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit in May 2016. The licensee reached its final compliance date on November 6, 2016, and has declared that the reactor is in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and proposed designs that if implemented appropriately should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders.

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COOPER NUCLEAR STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF MITIGATING STRATEGIES AND RELIABLE SPENT FUEL POOL INSTRUMENTATION RELATED TO ORDERS EA-12-049 AND EA-12-051 DATED SEPTEMBER 20, 2017

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