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

December 10, 2017

Mr. Christopher R. Church
Site Vice President
Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT– 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. MF0923 AND MF0924; EPID NOS. L-2013-JLD-0015 AND L-2013-JLD-0016)

Dear Mr. Church:

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. ML13066A066), Northern States Power Company, a Minnesota corporation (NSPM, the licensee), doing business as Xcel Energy, submitted its OIP for Monticello Nuclear Generating Plant (Monticello) in response to Order EA-12-049. At six month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated November 25, 2013 (ADAMS Accession No. ML13220A139), and March 17, 2015 (ADAMS Accession No. ML15072A098), the NRC issued an Interim Staff Evaluation (ISE) and audit report, respectively, on the licensee's progress. By letter dated July 6, 2017 (ADAMS Accession No. ML17187A153), NSPM 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 September 28, 2017 (ADAMS Accession No. ML17272A035), the licensee submitted a supplement to the FIP.

By letter dated February 28, 2013 (ADAMS Accession No. ML13060A447), the licensee submitted its OIP for Monticello 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 October 28, 2013 (ADAMS Accession No. ML13275A187), and March 17, 2015 (ADAMS Accession No. ML15072A098), 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 July 28, 2015 (ADAMS Accession No. ML15212A114), NSPM 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 NSPM's strategies for Monticello. The intent of the safety evaluation is to inform NSPM 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 Peter Bamford, Beyond-Design-Basis Management Branch, Monticello Project Manager, at 301-415-2833 or at Peter.Bamford@nrc.gov.

Sincerely,

Mohamed K. Shams, Chief
Beyond-Design-Basis Management Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No.: 50-263

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

NORTHERN STATES POWER COMPANY – MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

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

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

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

2.0 REGULATORY EVALUATION

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

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

2.1 Order EA-12-049

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

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

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

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

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

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

On December 10, 2015, 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," [Reference 6] 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, issued Japan Lessons-Learned Directorate (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," [Reference 7], 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, [Reference 5] requires that operating power reactor licensees and construction permit holders install reliable SFP level instrumentation. Specific requirements of the order are listed below:

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

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

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

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

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

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

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

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

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

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

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

On August 24, 2012, following several submittals and discussions in public meetings with NRC staff, the NEI submitted document NEI 12-02, "Industry Guidance for Compliance With NRC Order EA-12-051, To Modify Licenses With Regard to Reliable Spent Fuel Pool

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

3.0 TECHNICAL EVALUATION OF ORDER EA-12-049

By letter dated February 28, 2013 [Reference 10], Northern States Power Company, a Minnesota corporation (NSPM, the licensee), doing business as Xcel Energy, submitted an Overall Integrated Plan (OIP) for Monticello Nuclear Generating Plant (Monticello, or MNGP) in response to Order EA-12-049. By letters dated August 28, 2013 [Reference 11], February 28, 2014 [Reference 12], August 28, 2014 [Reference 13], February 24, 2015 [Reference 14], August 19, 2015 [Reference 15], February 22, 2016 [Reference 16], August 19, 2016 [Reference 17], and February 20, 2017 [Reference 18], the licensee submitted six-month updates to the OIP. In addition, by letter dated October 12, 2015 [Reference 19] the licensee submitted updated information regarding Order EA-12-049. By letter dated August 28, 2013 [Reference 20], the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 38]. By letters dated November 25, 2013 [Reference 21] and March 17, 2015 [Reference 22], the NRC issued an Interim Staff Evaluation (ISE) and an audit report on the licensee's progress. By letter dated July 6, 2017 [Reference 23], the licensee reported that full compliance with the requirements of Order EA-12-049 was achieved, and submitted a Final Integrated Plan (FIP). By letter dated September 28, 2017 [Reference 58], the licensee submitted a supplement to the 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.

Monticello is a General Electric (GE) boiling-water reactor (BWR), Model 3, with a Mark I containment. The licensee's three-phase approach to mitigate a postulated ELAP event, as described in the FIP, is summarized below. The approach is somewhat different if the plant receives warning of a pending Mississippi River flood that could result in water level rising above a pre-established level. In this case, the plant would be preemptively placed in cold shutdown, along with other preparatory actions.

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 (torus) located in the (primary) containment. Makeup to the RPV is provided primarily by the reactor core isolation cooling (RCIC) turbine-driven pump. Because the condensate storage tank (CST) is not robust for all postulated external events, the licensee's mitigating strategy assumes that the RCIC pump suction realigns to the suppression pool. The licensee expects the plant operators to take manual control of the SRVs to begin a controlled cooldown and depressurization of the reactor within 30 minutes of the event initiation. The cooldown is stopped when reactor pressure reaches a control band of 150 pounds per square inch gauge (psig) to 300 psig, to ensure sufficient steam pressure to operate the RCIC pump. When the primary containment pressure and suppression pool water temperature reach predetermined setpoints, the hardened containment wetwell vent (HCV, called the "hardened pipe vent" or HPV in the licensee's terminology) is opened to mitigate the primary containment temperature rise and allow the RCIC system to continue to function. The RPV makeup is expected to be provided from the RCIC system for as long as possible, since it is a closed loop system using relatively clean suppression pool water. When the RCIC system is no longer available or desired to be used, the RPV makeup supply comes from a diesel-driven FLEX pump.

The licensee performed a containment evaluation and determined that opening the suppression pool vent to atmosphere will allow containment temperature and pressure to stay within acceptable levels until supplemental equipment can be deployed to allow for restoration of the necessary plant systems.

The Monticello SFP is located in its Reactor Building. The pool will initially heat up due to the unavailability of the normal cooling system. During this time SFP level would be monitored by equipment installed to meet the requirements of Order EA-12-051. According to the licensee's FIP, SFP cooling will be initiated in Phase 2 via boiling and makeup. This will be accomplished by using a diesel-driven makeup pump (the same pump as is used for core cooling after use of the RCIC system is discontinued), along with the necessary hoses and/or installed piping systems to provide a makeup path. In addition, the Reactor Building will be vented to allow steam generated by boiling to exit the area of the SFP. The licensee has calculated that boiling could start as soon as 8.3 hours after the start of the ELAP assuming a full core offload, or within approximately 36 hours for a normal SFP heat load. The SFP makeup would be performed by using a diesel-driven portable FLEX pump supplied from the Mississippi River (the ultimate heat sink, or UHS), discharging to installed systems, or alternatively, directly to the SFP via hoses. Later in Phase 3, NSRC equipment would provide additional backup capability.

During the postulated event, the operators will complete a deep dc bus load shed within 2 hours of event initiation to ensure safety-related battery life is extended up to approximately 11 hours. Following dc load stripping and prior to battery depletion, a 200 kilowatt (kW), 480 volt alternating current (Vac) portable FLEX diesel generator (DG) will be deployed from a storage building. This DG will be used to repower essential 250 and 125 Volts-dc (Vdc) battery chargers, allowing the RCIC system, the SRVs, the HPV, and critical instrumentation to continue functioning.

To provide long-term support for indefinite coping, a National Strategic Alliance for FLEX Emergency Response (SAFER) Response Center (NSRC) will provide high capacity pumps and large combustion turbine-driven electrical generators (CTGs), which could be used to power one division of essential power to support continued cooling functions.

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. In Phase 1, 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 DGs 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 for the ELAP with loss of normal access to the UHS event 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 during the event. Maintenance of sufficient RPV inventory, while accommodating for 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 are discussed in further detail below. The licensee's strategy for ensuring compliance with Order EA-12-049 for conditions where the unit is shut down or being refueled is reviewed separately in Section 3.11 of this safety evaluation.

3.2.1 Core Cooling Strategy and RPV Makeup

3.2.1.1 Phase 1

According to the licensee's FIP, the injection of cooling water into the RPV will be accomplished through the RCIC system. Because the turbine for the RCIC pump is driven by steam from the RPV, operation of the RCIC system further assists the SRVs with RPV pressure control. The RCIC system suction is normally lined up to the CST. However, the CST is not protected from all applicable hazards and the licensee's FIP states that the RCIC pump suction will swap to the suppression pool with an assumed loss of the CST. If the CST survives the initiating external event intact, its inventory could be utilized; however the licensee's plan conservatively assumes that it would not be available.

Pressure control of the RPV is accomplished using the pneumatically-operated SRVs. The SRV controls are powered from of the 125 Vdc buses. Operators will utilize the SRV's to depressurize the RPV to 150-300 psig at a rate of less than 100 degrees Fahrenheit (°F) per hour. After this depressurization, the RPV pressure is maintained within the 150-300 psig control band to allow for continued operation of the RCIC system. There is a backup nitrogen system in place to allow the continued operation of the SRV's for at least 25 hours after the initiation of the ELAP event.

The station batteries and the Class 1E 125 Vdc distribution system provide power to RCIC systems and instrumentation. According to the sequence of events presented in the licensee's FIP, Table 10.0-1, dc load shedding is completed approximately 2 hours after the beginning of the event. This load shedding will extend the battery capacity to power the Phase 1 systems and instrumentation until the FLEX DG is deployed and connected in Phase 2, which should occur approximately 11 hours into the event.

3.2.1.2 Phase 2

The licensee's FIP states that RCIC will continue to be used until it is necessary to transfer to a portable FLEX pump. When RCIC is no longer available, or conditions in the suppression pool dictate, the strategy transitions to low pressure RPV makeup. This transition is accomplished by deploying a portable FLEX pump such that it takes a suction from the intake or discharge canal. The pump will discharge to the RPV via hoses connected to the "A" residual heat removal service water (RHRSW) system. An alternative connection is provided in the "A" residual heat removal (RHR) system. Since the primary and alternate connections are made on the same train, this configuration does not fully conform to the provisions of NEI 12-06, Revision 2, which specifies that primary and alternate connections should not be made in one division/train. The licensee has proposed the common-train RPV makeup connections as an alternative to NEI 12-06, and the alternative is evaluated in Section 3.14 of this safety evaluation.

During the postulated ELAP event, the SRVs would lose ac power and instrument air. The SRVs can be operated with dc power and a backup nitrogen supply. This ensures control of RPV pressure throughout the event.

According to the licensee's FIP, the usage of the intake and discharge canal will provide adequate volume of makeup water for the core cooling strategy into Phase 3. The licensee performed a hydraulic analysis to confirm that the portable pump will satisfy makeup requirements for both the core cooling and the SFP makeup strategies. The pumps and hydraulic calculations are discussed in Section 3.2.3.5 of this safety evaluation.

In the event that raw water is used to provide core cooling, the licensee's information update dated October 12, 2015 [Reference 19], indicates that Monticello will utilize the guidance contained in BWR Owners Group (BWROG) report BWROG-TP-14-006, "Fukushima Response Committee Raw Water Issue: Fuel Inlet Blockage from Debris." This guidance contains direction to maintain the RPV water level at the level of the moisture separator drains to ensure that core cooling is maintained while considering the possible clogging of fuel element orifices and filters.

3.2.1.3 Phase 3

According to the licensee's FIP, the Phase 3 strategy would be to maintain and supplement/replace the Phase 2 strategy with Phase 3 equipment. The Phase 3 equipment begins to arrive within 24 hours of the NSRC notification. It is then available to replace or supplement the Phase 2 components. According to the licensee's FIP, the site emergency response organization (ERO) would develop plans for deployment and priorities for restoration of systems.

3.2.2 Variations to Core Cooling Strategy for Flooding Event

For a flooding event, the licensee uses the expected warning time to bring the reactor to cold shutdown when the Mississippi River is predicted to exceed a pre-determined level. In addition the licensee pre-stages FLEX equipment, including the FLEX DG and portable pump, to a location that is protected from the flood waters. Since the CST is expected to be available during a flooding event, the licensee plans to connect hoses from the CST to the FLEX portable pump during the flood warning time. Thus, a cleaner source of RPV makeup would be available for core cooling in this scenario.

3.2.3 Staff Evaluations

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

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

3.2.3.1.1 Plant SSCs

Phase 1

In the licensee's FIP, Section 3.1.1 states that the primary strategy for core cooling is to supply high quality water via the RCIC system. 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 batteries is needed to operate valves and controls. In addition, the staff confirmed during the audit process that station procedure A.8-05.01, "Manual Operation of RCIC," Revision 9, directs operators to manually operate RCIC in the event that automatic operation is not available. As described in the licensee's FIP, Section 3.1.4.1, the RCIC system is located in the Reactor Building, which is robust with respect to a Safe Shutdown Earthquake (SSE), wind loading, tornado generated missiles, and all other postulated external events. In NEI 12-06, Section 3.2 further states that equipment designed to be robust is assumed to be fully available. In order to confirm the robustness of the RCIC system, the staff reviewed the licensee's evaluations conducted for the expedited seismic evaluation process (ESEP) during the audit process. These evaluations covered the RCIC system and its attendant instrumentation and controls. Further, the staff evaluated the licensee's ESEP report and documented its review in a letter dated October 14, 2015 [Reference 57]. Based on the licensee's FIP statements, supplemented by the staff's review of ESEP documentation during the audit process, the NRC staff finds the RCIC system is robust, and would be expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

The primary strategy for reactor pressure control and decay heat removal 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 to provide a motive force for operation. Sections 3.1.4.2 and 3.1.4.6 of the licensee's FIP state that the normal pneumatic supply would be unavailable with a loss of ac power and backup accumulators would supply a limited amount of pressure to support valve operation. As stated in the licensee's FIP, Section 3.1.4.6, the backup system will provide a pneumatic source for the SRVs for at least 25 hours after the start of the event. Subsequently, nitrogen bottles will be replaced to provide additional capability. The backup system is safety-related and located in the Reactor Building. The SRVs and controls are located in the primary containment which is protected from all applicable hazards. Additionally, the SRVs are safety-related. Based on the licensee's FIP statements, the NRC staff concludes that the SRVs and support systems 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. RCIC will continue to be used with suction from the suppression pool and the SRVs will continue to be used for pressure control. Availability of RCIC and the SRV's is discussed in the Phase 1 portion of this safety evaluation section above. In the FIP, Section 3.1.2 states that when RCIC operation is no longer possible, the RPV will be fully depressurized and a FLEX pump will be deployed which can inject river water into the RPV through primary or alternate connection points. The licensee plans to rely on the FLEX connection points and water sources discussed in Sections 3.7 and 3.10 of this safety evaluation, respectively, to accomplish this method of RPV makeup.

Phase 3

The licensee's Phase 3 strategy does not rely on any additional installed plant SSCs other than those discussed in Phase 1 and 2. However, the licensee will have the capability to power an existing safeguards 4160 Vac bus using equipment from the NSRC, thus providing shutdown cooling capability. This would occur under the direction of the ERO, and would be based on the existing plant conditions and availability of the necessary plant equipment.

3.2.3.1.2 Plant Instrumentation

The licensee's plan is to monitor instrumentation in the main control room and, as a backup, 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 FLEX power supply that supports this instrumentation is contained in Section 3.2.2.6 of this safety evaluation.

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

- RPV Level
- RPV Pressure
- Drywell Pressure

- Drywell Air Temperature
- Suppression Pool Level
- Suppression Pool Temperature

These instruments can be monitored from the control room, or locally at instrument racks. The staff notes that 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.

According to the licensee's FIP, instrumentation would be powered by station batteries during the ELAP event. Within 11 hours of the ELAP event initiation, a FLEX DG would be deployed which would provide power to the battery chargers through the emergency buses. The FLEX generators continue to provide power for the duration of the event. Additional backup generators will be available from NSRC during Phase 3. Therefore, since the instrumentation to implement the strategy is located in the control room and maintains power for all event phases, the NRC staff finds that the locations of the instrument indications should be accessible continuously throughout the ELAP event.

In accordance with NEI 12-06, Section 5.3.3.1, guidelines for obtaining critical parameters locally should be provided in an FLEX Support Guideline (FSG). During the audit process, the staff reviewed the licensee's guideline, A.8-06.04, "Alternate Methods for Monitoring RX Vessel and Containment Parameters," Revision 2, to confirm that it provides alternate methods for obtaining critical parameters if key parameter instrumentation is unavailable, consistent with the provisions of NEI 12-06.

3.2.3.2 Thermal-Hydraulic Analyses

The licensee concluded that its mitigating strategy for reactor core cooling would be adequate based in part on thermal-hydraulic analysis performed using Version 4 of the Modular Accident Analysis Program (MAAP). 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 for Monticello. This dependency notwithstanding, the NRC staff's discussion in this section focuses on the licensee's analysis of reactor core cooling. The NRC staff's review of the licensee's analysis of containment thermal-hydraulic behavior is provided in Section 3.4.4.2 of this safety evaluation.

The 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 Dai-ichi 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 Version 4 of the MAAP code (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 reviewed the capability of the MAAP4 code for performing thermal-hydraulic analysis of the ELAP event for both BWRs and PWRs. The NRC staff's review involved a limited review of key code models, as well as confirmatory evaluations 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 [Reference 24], 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 the Monticello Technical Specification 3.4.9 limits, were satisfied. Specifically, the licensee's analysis calculated that operators would maintain the collapsed liquid level in the reactor vessel 3 feet above the top of the active fuel region throughout the analyzed ELAP event. By maintaining the reactor core fully covered with water, adequate core cooling is assured for this event. Additionally, the licensee's fulfillment of the endorsement letter condition regarding the primary system cooldown rate at Monticello 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, that the licensee should be capable of maintaining 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, a FLEX pump) 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 makeup 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 Monticello assumed a total leakage rate at normal RPV operating pressure of 165 gallons per minute (gpm). This leakage rate used in the licensee's MAAP4 analysis was based on the failure of both seals (70 gpm per seal) plus 25 gpm primary system boundary leakage, consistent with Monticello Technical Specification 3.4.4. According to the licensee's FIP, the RCIC system or the FLEX pump will provide sufficient cooling flow to compensate for any reactor coolant lost in addition to maintaining sufficient coolant in the RPV.

The NRC staff concludes that the leakage rate of 70 gpm is reasonable based on a review of the content in NRC Generic Letter 91-07 [Reference 50]. Gross seal failures are not anticipated to occur during the postulated ELAP event. As is typical of the majority of U.S. BWRs, Monticello 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. The licensee's analysis shows that, at the limiting pressure, the FLEX pump is able to inject at a rate which maintains adequate margin.

Based upon 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 Monticello's Updated Safety Analysis Report (USAR) [Reference 55], Section 3.3.3.3, the control rods provide adequate shutdown margin under the core's maximum reactivity condition, with the assumption that the highest-worth control rod remains fully withdrawn. Monticello 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 acceptable shutdown margin for the analyzed ELAP event.

3.2.3.5 FLEX Pumps and Water Supplies

The licensee relies on one portable diesel driven pump during Phase 2. When RCIC is no longer available, the FLEX pump is used to inject river water into the RPV from primary or alternate connection points. As described in FIP Section 2.3.9, this pump takes suction from the intake canal, discharge canal, or CSTs, if available. The FLEX pump is rated at 1000 gpm at 455 feet of head.

To evaluate the capacity of the FLEX pump, the licensee performed hydraulic analysis 15-004, "Monticello Flex Pump Simultaneous SFP/RPV Flow," Revision 1. The staff reviewed the licensee's hydraulic analysis during the audit process and confirmed that the FLEX pump capacity was sufficient. During the onsite audit, the staff also 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 are consistent with the descriptions in the hydraulic analysis.

Section 3.1.8.1 of the licensee's FIP shows that two FLEX pumps are provided in order to satisfy the NEI 12-06 provision for "N+1" capability. The FLEX pump described in this section is also used for SFP makeup. The two FLEX pumps, along with other supporting equipment, are stored (one each) in the licensee's two FLEX storage facilities.

Based on the staff's review of the FLEX pumping capabilities at Monticello, as described in the above hydraulic analyses and the FIP, the NRC staff concludes that the portable FLEX pumps should perform as intended to support core cooling and RPV makeup during an ELAP 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 postulated events. The electrical strategies described in the FIP maintain the key safety functions of core cooling, containment and SFP cooling in an integrated manner and any exceptions between the key safety functions are noted in Sections 3.3.4.4 and 3.4.4.4 of this safety evaluation.

According to the licensee's FIP, operators would declare an ELAP following a loss of offsite power, emergency diesel generators (EDGs), and any alternate ac source. The plant's indefinite coping capability is attained through the implementation of pre-determined FLEX strategies that are focused on maintaining or restoring key plant safety functions.

During the first phase of an ELAP event, Monticello would rely on the safety-related Class 1E batteries to provide power to key instrumentation and applicable dc components. The station batteries and associated dc distribution systems are located within safety-related structures designed to meet applicable design-basis external hazards. In order to confirm the timeline described in the licensee's FIP, the NRC staff reviewed procedure C.5-4401, "FLEX DC Load Shed," Revision 3, during the audit process. This procedure directs 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 is available in Phase 2. The plant operators would commence load shedding within 1 hour and complete load shedding within 2 hours from the onset of the postulated event.

Monticello has four Class 1E station batteries that are utilized in the implementation of the FLEX strategy. Specifically, these are described as two 125 Vdc (D1 and D2) and two 250 Vdc (D3 and D6) station batteries. Station batteries D1 and D2 are model KCR-13 with a capacity of 491 ampere-hours (AH). Station batteries D3 and D6 are model KCR-19 with a capacity of 697 AH and 573 AH, respectively. According to the licensee's FIP, Monticello also has a non-Class 1E 250 Vdc station battery (D7) that is used to support communications equipment. Station battery D7 is model XT1L-29 with a capacity of 1459 AH. All of these batteries were manufactured by C&D Technologies. The battery coping time for batteries D1, D2, D3, D6, and D7 is 12 hours.

The NEI White Paper, "EA-12-049 Mitigating Strategies Resolution of Extended Battery Duty Cycles Generic Concern" [Reference 40], provides guidance for calculating extended duty cycles of batteries (i.e., beyond 8 hours). This paper was endorsed by the NRC [Reference 41]. 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 [Reference 42]. The testing provided additional validation that the NEI white paper method was technically acceptable. During the audit process, the NRC staff reviewed the licensee's battery calculations and confirmed that they had followed the guidance in the white paper. Specifically, the NRC staff reviewed the licensee's dc coping calculations 14-099, "125V D1 Battery FLEX Coping Time Analysis," Revision 1, 14-100, "125V D2 Battery FLEX Coping Time Analysis," Revision 1, 14-101, "250V D3 Battery FLEX Coping Time Analysis," Revision 0, 14-102, "250V D6 Battery FLEX Coping Time Analysis," Revision 0, and 14-036, "250V D7 Battery FLEX Coping Time Analysis," Revision 1, which verified the capability of the dc system to supply power to the required loads during the first phase of the Monticello FLEX mitigation strategy plan. 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 2 hours to extend battery capability up to 12 hours.

Based on the staff's review of the licensee's FIP statements, confirmed by reviewing the licensee's analyses and procedures, and the battery vendor's capacity and discharge rates for the Class 1E station batteries, the NRC staff finds that the licensee's dc systems have adequate capacity and capability to power the loads required to mitigate the consequences during Phase 1 of an ELAP at Monticello, provided that necessary load shedding is completed within the times assumed in the licensee's analyses.

The licensee's Phase 2 strategy includes repowering the Class 1E battery chargers within 11 hours after initiation of an ELAP to maintain availability of instrumentation to monitor key parameters. This strategy relies on portable 200 kW 480 Vac FLEX DGs. The licensee has two of these FLEX DGs, with only one being required to execute the strategy. The 480 Vac FLEX DG would provide power to 125/250 Vdc battery chargers and provide continuous dc power for the SRV's, the RCIC system, the HPV system, and other selected loads.

During the audit process, the NRC staff reviewed licensee evaluation EC [Engineering Change] 23964, "FLEX 480 V Diesel Generator Sizing," Revision 0, conceptual single line diagrams, and the separation and isolation of the FLEX DGs from the EDGs. The evaluation shows that the minimum loading for the licensee's Phase 2 200 kW FLEX DG is 163.8 kW. The staff review noted that the licensee evaluation took the FLEX cable lengths into consideration when evaluating the minimum voltage at the limiting component. Based on its review of the licensee's calculation, the NRC staff finds that the 200 kW FLEX DG is adequately sized to support the electrical loads required for the licensee's Phase 2 strategies, as described in the FIP.

For Phase 3, the licensee plans to continue the Phase 2 coping strategy with additional assistance provided from offsite equipment/resources. The offsite resources that will be provided by an NSRC includes two 1-megawatt (MW) 4160 Vac CTGs, one 1100 kW 480 Vac CTG, and distribution panels (including cables and connectors). According to licensee calculation 92-224, "Emergency Diesel Generator Loading," Revision 5D, the loads to be powered by the Phase 3 CTGs would total 1353.7 kW and 1462.2 kW depending on the bus (train) chosen for energization. These loads fall within the rating of the 4160 Vac Phase 3 CTGs. The Phase 3 4160 Vac CTGs would supply power for one RHR pump, one RHRSW pump, related valves, and other miscellaneous loads. Based on its review, the NRC staff finds that the equipment being supplied from either of the NSRCs has sufficient capacity and capability to supply the required loads during Phase 3.

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

3.2.4 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed guidance that should maintain or restore core cooling and RPV inventory during an ELAP event consistent with NEI 12-06 guidance, as endorsed by JLD-ISG-2012-01, considering one alternative that is discussed in Section 3.14 of this safety evaluation, and should adequately address the requirements of the order.

3.3 Spent Fuel Pool Cooling Strategies

In NEI 12-06, Revision 2, 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 spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design-basis heat load. 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.

However, in JLD-ISG-2012-01, Revision 1 [Reference 7], 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. This evaluation must use the reevaluated seismic hazard that is described in Section 3.5.1 of this safety evaluation if its magnitude is higher than the site's SSE.

Regarding the SFP integrity evaluation, as part of the process to develop and evaluate responses to an NRC Request for Information issued pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f) [Reference 25] (hereafter referred to as the 50.54(f) letter), the NRC staff worked with industry to develop a method of predicting the susceptibility of a SFP to cracking. The primary input considers the seismic stresses that the SFP was designed to survive compared to the seismic stresses from a seismic event that has some probability of occurring at the site. By letter dated February 23, 2016 [Reference 45], NEI submitted a request for the NRC staff to approve guidance for a SFP evaluation based on the

February 2016 EPRI report 3002007148, "Seismic Evaluation Guidance Spent Fuel Pool Integrity Evaluation." The EPRI report is a generic study performed for nuclear power plants with low-to-moderate seismic ground motions (peak spectral accelerations less than 0.8g) to address the NRC 50.54(f) letter provision requesting an evaluation of the SFP that considers all seismically induced failures that could lead to rapid draining. Based on the NRC staff's assessment letter dated July 8, 2015 [Reference 47], the peak spectral acceleration for the licensee's SSE and the reevaluated ground motion response spectra are less than 0.8g at Monticello; thus, the NRC staff finds that EPRI 3002007148 is applicable to the licensee's site. Section 3.3 of EPRI 3002007148 provides guidance on specific site parameters, structural parameters, and nonstructural parameters that a licensee should confirm on a site-specific basis to ensure that the report's conclusions apply. By letter dated March 17, 2016 [Reference 46] the NRC staff endorsed the EPRI report as an acceptable method of performing seismic evaluations of SFPs in responding to the 50.54(f) letter. By letter dated October 28, 2016 [Reference 48], the licensee confirmed that the Monticello SFP met the parameters in Section 3.3 of the EPRI report to affirm that the report was applicable to the site and to confirm that the Monticello SFP was seismically adequate in accordance with the NTTF recommendation 2.1 seismic evaluation criteria. By letter dated December 8, 2016 [Reference 49], the NRC staff concluded that the licensee responded appropriately to the 50.54(f) letter regarding the SFP evaluation. Specifically, the NRC review noted that the licensee's submittal: (1) appropriately evaluated and screened the SFP SSCs; and demonstrated that the Monticello SFP structure is sufficiently robust and can withstand ground motions with peak spectral acceleration less than or equal to 0.8g, and (2) adequately evaluated the non-structural considerations of the Monticello SFP whose failure could lead to potential drain-down due to a seismic event.

Based on the licensee's evaluation and demonstration of conformance to the EPRI 3002007148, Section 3.3 criteria; the NRC staff concludes that the licensee has demonstrated that SFP spray flow is not needed in the licensee's mitigating strategy, consistent with JLD-ISG-2012-01, Revision 1.

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 and, 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 staff's evaluation presented in this safety evaluation section primarily addresses the anticipated response during non-full core offload scenarios. The effects of a postulated event with a full core offload is addressed in Section 3.11 of this safety evaluation.

3.3.1 Phase 1

Regarding Phase 1 of the postulated event, the licensee's FIP states that no actions are required for SFP makeup because the time to boil is sufficient to allow for deployment of Phase 2 equipment. Adequate SFP inventory exists to provide radiation shielding for personnel well beyond the time of boiling. During this phase the licensee plans to monitor SFP water level using reliable SFP level instrumentation installed per Order EA-12-051.

3.3.2 Phase 2

In the licensee's FIP, Section 3.2.2 states that during Phase 2, operators will deploy a portable FLEX pump to supply water from either the discharge canal or the intake to provide makeup water to the SFP. This portable pump is the same pump as is used for RPV makeup. The licensee would also have the option of using CST water, if it were to be available. The options for routing the FLEX pump discharge hoses would be: (1) to a connection in the RHRSW system that aligns to the fire header and would then allow makeup deployment from one of seven fire hose stations on the refueling floor, (2) directly to the refueling floor to provide makeup to the SFP, or (3) to the same RHRSW connection as option (1), with alignment to the SFP cooling system (not requiring refueling floor access).

3.3.3 Phase 3

The FIP states that SFP cooling can be maintained indefinitely using the same makeup strategies as in Phase 2. However, NSRC equipment is available during Phase 3 to provide additional defense-in-depth.

3.3.4 Staff Evaluations

3.3.4.1 Availability of Structures, Systems, and Components

3.3.4.1.1 Plant SSCs

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

The licensee's FIP indicates that boiling could begin at approximately 36.9 hours during a normal, non-outage situation. The staff noted that the licensee's sequence of events timeline in the FIP indicates that operators will initiate makeup to the SFP prior to 57 hours into the event to ensure the SFP area remains habitable for personnel entry. According to the licensee's FIP, the SFP level would drop less than 2 feet by this time, assuming a normal SFP heat load. The licensee's Phase 1 SFP cooling strategy does not require any operator actions. However, the licensee does establish a ventilation path to cope with temperature, humidity and condensation from evaporation and/or boiling of the SFP. The operators are directed to open the rail bay door and personnel doors in the Reactor Building to establish the ventilation path. During the audit process, the NRC staff reviewed licensee procedure C.4-B.09.02.A, "Station Blackout," Revision 46. Step 30 of this procedure directs operators to establish a natural convection air flow path in accordance with C.5-4301, "Reactor Building Ventilation during FLEX Conditions," Revision 2. Setup of SFP makeup activities is initiated by Step 43 of C.4-B.09.02.A, after natural ventilation pathways have been established. The staff confirmed that procedure C.5-4301, "Spent Fuel Pool Makeup with FLEX Portable Diesel Pump", Revision 2, directs operators to complete setup activities prior to the occurrence of boiling in the SFP because of habitability concerns.

The licensee's Phase 2 and Phase 3 SFP cooling strategy involves the use of the FLEX pump (or possibly the NSRC-supplied pump for Phase 3), with suction from the UHS, to supply water to the SFP. The staff's evaluation of the robustness and availability of FLEX connections points for the FLEX pump is discussed in Section 3.7.3.1 of this safety evaluation. Furthermore, the staff's evaluation of the robustness and availability of the UHS for an ELAP event is discussed in Section 3.10.3.

3.3.4.1.2 Plant Instrumentation

In its FIP, the licensee stated that the instrumentation for SFP level will meet the requirements of Order EA-12-051. Furthermore, the licensee stated that these instruments will have initial local battery power with the capability to be powered from the FLEX DGs. The NRC staff's review of the SFP level instrumentation is discussed in Section 4 of this safety evaluation.

3.3.4.2 Thermal-Hydraulic Analyses

As described in Section 3.2.6 of this safety evaluation, the SFP will boil in approximately 8.3 hours and boil off to a level to top of fuel in 63.2 hours from initiation of the event with no operator action at the maximum design heat load as described in the Monticello USAR. The heat load for this evaluation corresponds to a full core offload. The USAR and FIP further state that a makeup rate of 53 gpm would be sufficient to compensate for boiling, assuming this heat load. The licensee conservatively evaluated a SFP makeup flow rate of at least 200 gpm (concurrent with 300 gpm of RPV makeup) in the supporting hydraulic analysis to show that adequate SFP level can be maintained. Consistent with the guidance in NEI 12-06, Section 3.2.1.6, the staff finds the licensee has considered the maximum design-basis SFP heat load. With a normal SFP heat load, the required makeup rate to maintain level in the SFP would be considerably less than is assumed in the hydraulic analysis, approximately 12 gpm.

3.3.4.3 FLEX Pumps and Water Supplies

As described in the FIP, the SFP cooling strategy relies on a FLEX pump to provide SFP makeup during Phase 2. In the FIP, Section 3.2.7.1 describes the hydraulic performance criteria (e.g., flow rate, discharge pressure) for the FLEX Pump. The FLEX pump is the same pump used for core cooling, and the licensee's analysis reflected simultaneous makeup to the RPV and SFP. The staff also reviewed the performance criteria of the pumps that would be supplied from an NSRC, and confirmed that the capability exists to perform the function of the onsite FLEX pump if it were to fail. Based on the review of the licensee's FIP, supplemented by a review of the SFP heat load/boil-off calculations and overall hydraulic analysis, the staff concludes that the Phase 2 and 3 pumps can provide adequate makeup flow to maintain the SFP cooling safety function.

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 and the capability of this instrumentation is described in Section 4 of this safety evaluation. Based on the review of this instrumentation in Section 4 of this safety evaluation, the NRC staff finds that the licensee's electrical strategy should allow the licensee to restore or maintain the SFP cooling function during ELAP conditions.

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 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.

According to the licensee's FIP, the maximum analyzed containment pressure under the postulated conditions is approximately 38 pounds per square inch absolute (psia) (approximately 23 psig). This is well below the Primary Containment Pressure Limit (PCPL) of 62 psig. During the audit process the NRC staff reviewed the licensee's supporting calculation, PRA-CALC-14-003, "Monticello MAAP Thermal Hydraulic Calculations to Support Extended Loss of AC Power (ELAP) Mitigating Strategies," Revision 1. The staff was able to confirm that this calculation was based on the boundary conditions described in Section 2 of NEI 12-06. The calculation analyzed the strategy removing reactor decay heat from the torus by venting steam from the suppression pool to the environment through the HPV and concluded that the containment parameters of pressure and temperature remain well below the respective USAR Section 5.2.1 design limits of 62 psig and 281°F for more than 24 hours. The evaluation modeled the first 24 hours, but demonstrated containment parameters can be maintained below limits beyond 24 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.

3.4.1 Phase 1

According to the licensee's FIP, the Phase 1 coping strategy for containment involves verifying containment isolation, and monitoring containment temperature and pressure using installed instrumentation. During the audit process, the staff reviewed Abnormal Procedure C.4-B.09.02.A, "Station Blackout," Revision 46, which provides guidance for verifying containment isolation. The staff also reviewed procedure A.8-06.04, "Alternate Methods for Monitoring RX Vessel and Containment Parameters," Revision 2, which provides guidance for monitoring primary containment temperature.

According to the licensee's FIP, as the reactor core decay heat is rejected to the suppression pool, the temperature and pressure in the torus will increase. Maintaining containment integrity will be achieved by the removal of energy from the torus by opening the HPV. The vent will be operated in accordance with the Emergency Operating Procedures (EOPs). The vent will be opened after the water in the suppression pool reaches 212°F and containment pressure is approximately 10 psig. This occurs approximately 7 hours after the ELAP event begins. In order to confirm the licensee's FIP statements, the staff reviewed procedure C.5-4000, "Station Blackout Guideline", Revision 1, which directs operators to vent primary containment when suppression pool water temperature exceeds 212°F and primary containment pressure is greater than 10 psig.

Once the HPV is opened, the suppression pool water temperature will peak at approximately 250°F and will slowly decrease over the next several days as decay heat decreases. After the release of the nitrogen overpressure, the containment pressure will rise to the saturation pressure associated with 250°F (approximately 30 psig), and then decrease as the energy removed from the containment exceeds the energy added. After 48 hours, the containment temperature will have lowered to approximately 240°F. According to the licensee's FIP, monitoring of containment drywell pressure, suppression pool level, drywell temperature, and suppression pool temperature will be available via normal plant instrumentation.

3.4.2 Phase 2

The Phase 2 strategy will continue torus venting through the HPV to maintain drywell and suppression pool temperatures and pressures within allowable limits. This is a continuation of Phase 1 strategy with the addition of FLEX electrical power to supply instruments identified for monitoring containment integrity, along with powering the HPV components and providing for supplemental nitrogen gas for operation of HPV valves. The licensee has no defined end time for the Phase 1 and 2 at Monticello with regards to maintaining containment integrity.

3.4.3 Phase 3

Monticello does not have pre-planned actions to utilize off-site resources for Phase 3. Phase 2 equipment is able to remove decay heat past 24 hours while maintaining containment integrity. The HPV line will be used to remove heat until off-site resources provided by the NSRC arrive on site. According to the licensee's FIP, the ERO will review the availability of plant equipment along with off-site resources and will develop plans to restore plant systems for the removal of heat from containment.

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

Hard Pipe Vent (HPV)

FIP Section 3.1.4.4, describes the HPV System as starting from an 8 inch pipe off the top of the torus. Two air operated valves provide containment isolation. The air operated valves are air to open and spring to close. Air is provided by a dedicated pneumatic system via dc solenoid valves. The dc solenoid valves are powered from dedicated batteries. Downstream of the containment isolation valves, the pipe increases to a 10 inch diameter. A rupture disc provides secondary containment isolation. Another dedicated pneumatic system is used to rupture the rupture disc if venting is desired. The vent line exits the plant buildings, up the outside of the Reactor Building to a point above the top of the building. The portion of the HPV system up to 30 feet above grade is protected from tornado missiles. Controls and indication for the HPV are located on or near the alternate shutdown system (ASDS) panel.

In the USAR, Section 5.2 indicates the HPV system is designed to prevent containment pressure from increasing under conditions of constant heat input at a rate equal to 1 percent of rated thermal power and containment pressure equal to the PCPL. The HPV system was initially installed in response to Generic Letter 89-16 and has been upgraded to a reliable, severe accident capable, wetwell venting system that complies with the Phase 1 requirements of Order EA-13-109, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions."

The HPV system is used to vent non-condensable gases and steam out of the torus to the atmosphere. Procedures direct the operator to begin the containment venting when the containment pressure is greater than 10 psig and the suppression pool temperature is greater than 212°F. The venting is isolated if drywell pressure drops to 5 psig and is repeated as necessary throughout the ELAP event.

During the audit, the licensee provided EC 26083, "Hardened Containment Venting System NRC Order EA-13-109 Phase 1," for the staff to review. This EC stated that the HPV is designed to be Seismic Class I as defined by the Monticello USAR, Section 12.02, which states all Class I equipment is designed to withstand the design basis earthquake (SSE). The majority of the HPV system is located in the Reactor Building so it is protected from all other applicable hazards. The EC 26083 also states that exterior portion of the vent is designed for tornado/wind loads without failure and is missile protected up to 30 feet in accordance. Therefore, the staff concludes that the HPV system is expected to be available at the start of an ELAP event consistent with NEI 12-06, Section 3.2.1.3.

Containment

As described in Section 3.1 of this safety evaluation, the primary containment at Monticello is a GE Mark I design. In this design, the primary containment system consists of a steel lightbulb-shaped drywell, a steel doughnut-shaped pressure suppression chamber, and interconnecting vent pipes. This provides the first containment barrier surrounding the reactor vessel and reactor primary system. Any leakage from the primary containment system is to the secondary containment system which consists of the Reactor Building, the plant standby gas treatment system, and the plant main stack.

The primary containment system (drywell and pressure suppression chamber) was designed to meet the following criteria:

Maximum Pressure:	62 psig
Internal Design Pressure:	56 psig
External Design Pressure:	2 psig
Design Temperature:	281°F

According to the Monticello USAR, the components of the primary containment system (drywell, vents, torus, and penetrations) are designed and constructed as Class I structures. Class I structures and equipment are those whose failure could cause significant release of radioactivity or which are vital to safe shutdown of the plant under normal or accident conditions and the removal of decay and sensible heat from the reactor. The design of Class I structures and equipment takes into account postulated environmental and accident loading. All Class I structures and equipment were analyzed to assure that a safe shutdown can be made after a design basis earthquake. Based on the USAR description, the staff concludes that the primary containment system satisfies the NEI 12-06 definition of robust.

3.4.4.1.2 Plant Instrumentation

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

The FIP credits the following Instrumentation for providing the following key parameters for all phases of the containment integrity strategy:

- Drywell Pressure: PT-7251A/B – Instrumentation is available in the control room and the ASDS panel.
- Torus Level: LT-7338A/B – Instrumentation is available in the control room and the ASDS panel.
- Drywell Air Temperature: TE-4247 A1/A2/B1/B2/C1/C2/D1/D2/E1/E2/F1/F2/ G1/G2/H1/H2 through TR 23-115 – Instrumentation is available in the control room.
- HPV Radiation Monitor: RE/RM-4544 – Instrumentation is available at the ASDS panel.
- HPV System Valve Position Indication: AO-4539 and AO-4540 - Instrumentation is available at the ASDS panel.
- HPV temperature: Vent temperature readout on RR-4544 at the ASDS panel.

Based on this information, the staff concludes that the licensee should have the ability to appropriately monitor the key containment parameters as delineated in NEI 12-06, Table 3-1.

3.4.4.2 Thermal-Hydraulic Analyses

The licensee's thermal-hydraulic analysis for containment is summarized in the FIP. During the audit process the staff reviewed the supporting calculation, PRA-CALC-14-003, and confirmed that it was performed based on the boundary conditions described in Section 2 of NEI 12-06. The analysis used the MAAP4 computer program to perform numeric computations of the fundamental thermodynamic equations which predict the heat up and pressurization of the containment atmosphere under ELAP conditions. The analysis assumes the unit is initially operating at full power (2004 megawatts-thermal). Initial suppression pool temperature is 80°F with an initial containment (drywell and torus) pressure of 15.45 psia. Reactor coolant system leakage, including reactor recirculation pump seals, is assumed to be 165 gpm. The RCIC pump flow is assumed to be 400 gpm for 2 hours and then 300 gpm for the remainder of the model. The analysis modeled 24 hours of coping strategy and after 24 hours it assumes conditions are stabilized and pressures and temperatures will decrease due to decreasing reactor decay heat.

Calculation PRA-CALC-14-003, Appendix F, addresses the scenario where the containment is vented through the torus HPV when the containment pressure exceeds 10 psig and the suppression pool is equal to or greater than 212°F. Under this scenario containment venting starts at roughly 7 hours after the start of the ELAP. At approximately 11.5-hours, suppression pool reaches 250°F, at which time RCIC is assumed to fail due to high suction water temperature. At this time the Phase 2 portable FLEX pump is used to replace the RCIC pump function. Makeup from the Mississippi River thus starts at approximately 11.5 hours. Peak containment pressure occurs at 7 hours and stays below the design pressure of 56 psig (70.7 psia). The maximum drywell temperature reaches 280°F and the maximum torus gas space temperature is approximately 270°F. Both remain below the design temperature limit of 281°F.

3.4.4.3 FLEX Pumps and Water Supplies

The Phase 2 strategy uses the diesel-driven FLEX Pump to supply water to the RPV. The heat from the RPV is transferred to the containment by opening the SRVs and is removed through opening the HPV system. Section 3.2.3.5 of this safety evaluation provides a description of the FLEX pump and water supplies.

In Phase 3 the containment pressure and temperature monitoring remains unchanged. However, after NSRC delivery and setup of supplemental 4160 Vac CTGs, additional cooling options for containment may be utilized. For example, an RHR pump and RHRSW pump may be started for torus cooling. In addition, containment cooling through starting of containment fans and a cooling water source may also be initiated. Due to the multitude of possible damage scenarios and the availability of plant equipment, the licensee's FIP states that pre-planning of recovery actions for Phase 3 prior to the event is not practical and the licensee relies on the ERO to develop plans for deployment and to set priorities for restoration of plant systems.

3.4.4.4 Electrical Analyses

The licensee's Phase 1 coping strategy is to monitor containment pressure and temperature using installed instrumentation, and maintain containment integrity using normal design features of the containment, such as the containment isolation valves and the HPV system. The licensee's strategy to repower instrumentation using the Class 1E station batteries is identical to what was described in Section 3.2.3.6 of this safety evaluation and is adequate to ensure continued containment monitoring. The installed HPV system has a dedicated 125 Vdc battery.

The dedicated 125 Vdc battery would supply power for a minimum of 24 hours to the HPV isolation solenoid valves, rupture disk solenoid valves, as well as a number of indications and alarms. The staff reviewed licensee calculation 16-006, "Hard Pipe Vent D8 Battery HCVS 125VDC Battery Calculation," Revision 1, during the audit process and confirmed that it evaluated the battery/battery charger sizing and device terminal voltages for the 125 Vdc HPV system. The results of the calculation showed that the 125 Vdc HPV battery is adequately sized to supply power to the HPV system for 24 hours following an ELAP.

The licensee's Phase 2 coping strategy is to continue monitoring containment pressure and temperature using installed instrumentation and maintaining containment integrity. The licensee's strategy to repower instrumentation using a 480 Vac, 200 kW FLEX DG is identical to what was described in Section 3.2.3.6 of this safety evaluation and is adequate to ensure continued containment monitoring. The licensee would transition to Phase 2 prior to depleting the HPV battery (i.e., within 24 hours). During the audit process, the staff confirmed that procedure C.5-4453, "Energize Hard Pipe Vent During SBO [station blackout]," Revision 0, provides guidance to transfer electrical power for HPV loads from battery D8 (125 Vdc HPV battery) to the Division II, 250 Vdc battery.

The licensee's Phase 3 strategy is to continue its Phase 2 strategy. Monticello 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 finds that it is reasonable to expect that the licensee could utilize these resources to supply power to the HPV components to maintain the containment safety function indefinitely.

Based on its review, the NRC staff concludes that the electrical equipment available onsite (e.g., Class 1E batteries, HPV battery, and 200 kW FLEX DGs) supplemented with the equipment that will be supplied from an NSRC (e.g., 4160 Vac CTGs and a 480 Vac CTG), there is sufficient capacity and capability to supply the required loads to maintain containment.

3.4.5 Conclusions

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

3.5 Characterization of External Hazards

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

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

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

References to external hazards within the licensee's mitigating strategies and this 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, the NRC staff issued a 50.54(f) letter, which requested that licensees reevaluate the seismic and flooding hazards at their sites using updated hazard information and current regulatory guidance and methodologies. Due to the time needed to reevaluate the hazards, and for the NRC to review and approve them, the reevaluated hazards were generally not available until after the mitigation strategies had been developed. The NRC staff has developed a proposed rule, titled "Mitigation of Beyond-Design-Basis Events," hereafter called the MBDBE rule, which was published for comment in the *Federal Register* on November 13, 2015 [Reference 54]. The proposed MBDBE rule would make the intent of Orders EA-12-049 and EA-12-051 generically applicable to all present and future power reactor licensees, while also requiring that licensees consider the reevaluated hazard information developed in response to the 50.54(f) letter.

The NRC staff requested Commission guidance related to the relationship between the reevaluated flooding hazards provided in response to the 50.54(f) letter and the requirements for Order EA-12-049 and the MBDBE rulemaking (see COMSECY-14-0037, Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluation of Flooding Hazards" [Reference 51]. The Commission provided guidance in an SRM to COMSECY-14-0037 [Reference 26]. The Commission approved the staff's recommendations that licensees would need to address the reevaluated flooding hazards within their mitigating strategies for BDBEEs, and that licensees may need to address some specific flooding scenarios that could significantly impact the power plant site by developing scenario-specific mitigating strategies, possibly including unconventional measures, to prevent fuel damage in reactor cores or SFPs. The NRC staff did not request that the Commission consider making a requirement for mitigating strategies capable of addressing the reevaluated flooding hazards be immediately imposed, and the Commission did not require immediate imposition. In a letter to licensees dated September 1, 2015 [Reference 39], the NRC staff 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 [Reference 6]. The NRC staff endorsed Revision 2 of NEI 12-06 in JLD-ISG-2012-01, Revision 1 [Reference 7]. 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. The licensee has submitted MSAs for both seismic [Reference 59] and flooding [Reference 60] at Monticello. The NRC staff has reviewed both of these MSAs and issued corresponding assessment letters [References 61 and 62 for seismic and flooding, respectively].

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 described the current design-basis seismic hazard, the design-basis or maximum earthquake. For the purposes of this safety evaluation the staff will use the current NRC term corresponding to this earthquake level, the SSE. According to the FIP, the SSE corresponds to an acceleration level of 0.12g. As described in the Monticello USAR [Reference 55], Sections 12.2.1.4 and 12.2.1.9, all Class I structures and equipment were analyzed to assure that a safe shutdown can be made during ground accelerations at 0.12g (design-basis or maximum earthquake). This acceleration of up to 0.12g corresponds to a horizontal ground acceleration. For the design of Class I structures and equipment the maximum horizontal acceleration and the maximum vertical acceleration were considered simultaneously. The vertical acceleration was taken as two thirds (2/3) of the horizontal ground acceleration, corresponding to 0.08g. 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

As described in the Monticello USAR, Section 2.4.1, the current design basis flood is described as a Probable Maximum Flood (PMF) event on the Mississippi River. The PMF is described in the FIP as a slowly developing event, thus allowing time to implement compensatory and protective measures before the flood waters inundate the site. The PMF would result in a peak water elevation of approximately 939 feet mean sea level (MSL) at the site location, which is above the plant grade elevation of 930 feet MSL. In this scenario, the flood waters are predicted to exceed the plant grade level for up to 11 days. During the time that the river is rising, estimated in USAR Section 12.2.1.7.1 at 12 days, the licensee plans to construct a berm that will protect the critical plant structures from the flood waters. Since both of the storage buildings housing FLEX equipment are outside of the area protected by the berm, the licensee would pre-deploy equipment from both buildings to a location inside the berm-protected zone. In this flood scenario, the plant access road and Staging Area "B" may also be inaccessible, and thus the licensee plans to construct a temporary access road connecting County Road 75 to the berm-protected area of the site along higher ground during the warning period. This access road would be built on owner controlled property. In addition, the licensee's flood procedures shut the plant down at a pre-defined river level. The licensee's procedures will also initiate deployment of NSRC equipment at a pre-established predicted river level. According to the licensee's FIP, the effects of groundwater ingress were discussed in the flooding hazard reevaluation report that was prepared in response to the Fukushima 50.54(f) letter. To confirm that groundwater effects would be manageable, the staff used the audit process to verify that under flooding conditions the licensee's procedure A.6, "Acts of Nature," Revision 56, contains steps to direct mitigation of groundwater intrusion. The staff also confirmed that procedure 8300-02, "External Flooding protection Implementation to Support A.6 Acts of Nature," Revision 7, and its supporting work orders, provides steps for the use of portable electrical sump pumps powered by portable generators to remove water from critical plant locations.

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

3.5.3 High Winds

In NEI 12-06, Section 7, provides the NRC-endorsed screening process for evaluation of high wind hazards. This screening process considers the hazard due to hurricanes and tornadoes. The screening for high wind hazards associated with hurricanes should be accomplished by comparing the site location to NEI 12-06, Figure 7-1 (Figure 3-1 of U.S. NRC, "Technical Basis for Regulatory Guidance on Design Basis Hurricane Wind Speeds for Nuclear Power Plants," NUREG/CR-7005, December, 2009); if the resulting frequency of recurrence of hurricanes with wind speeds in excess of 130 miles per hour (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 site is located at 45° 20' North latitude and 93° 50' West longitude. Based on this location, NEI 12-06 Figure 7-2, Recommended Tornado Design Wind Speeds for the 1E-6/year Probability Level indicates the site is in a region where the tornado design wind speed exceeds 130 mph. Therefore, the plant screens in for an assessment for high winds and tornados, including missiles produced by these events. The NRC staff notes that the site is beyond the range of high winds from a hurricane per NEI 12-06, Figure 7-1. Thus, the staff concludes that a hurricane hazard is not applicable at Monticello and need not be addressed.

The Monticello USAR, Section 12.2.1.8, describes the design-basis loading conditions for tornado events. These are as follows:

- A rotational wind having a tangential velocity of 300 mph.
- Differential pressure between inside and outside enclosed areas - 2 psi.
- A torsional moment resulting from applying the wind specified in (a) above, only on one-half of the structure.

In addition, those areas housing critical equipment required to assure safe shutdown of the reactor were designed to prevent penetration of exterior walls from the following two types of missiles that could be generated by a tornado:

- A utility pole 35-feet long by 14-inches in diameter and a unit weight of 35 lbs per cubic foot having a velocity of 200 mph.
- A 1 ton missile, such as a compact type automobile traveling at 100 mph at a maximum height of 25-feet above grade and with a contact area of 25 square feet.

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 NEI 12-06, Figure 8-2, should address the impact of ice storms.

In its FIP, regarding the determination of applicable extreme external hazards, the licensee stated that the site is located 45° 20' North latitude and 93° 50' West longitude. Based on this location, the site is located within the region characterized by EPRI as ice severity Level 4 (NEI 12-06, Figure 8-2, Maximum Ice Storm Severity Maps). Consequently, the site is subject to icing conditions that could cause severe damage to electrical transmission lines and/or accumulations of a large amount of ice. The licensee concludes that the plant screens in for an assessment for snow, ice, and extreme cold hazard. In its FIP, the licensee stated that FLEX equipment is protected from severe temperatures.

Regarding snowfall and extreme cold, the Monticello USAR, Section 2.3, describes the average annual snowfall for the region of Minnesota where the site is located as approximately 42 inches per year, with a historical variance from 6 inches to 88 inches. The maximum snowfall in a 24 hour period for Minneapolis was approximately 16 inches. Based on 54 years of recorded data, the USAR describes an extreme cold temperature in the area as -38°F.

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, as per NEI 12-06 Section 9.2, all sites are required to consider the impact of extreme high temperatures. Summers at the site may bring periods of extremely hot weather, with the USAR, Section 2.3 describing a historical observed maximum of 107°F. Thus, the plant site screens in for an assessment for extreme high temperature hazard.

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

3.5.6 Conclusions

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

3.6 Planned Protection of FLEX Equipment

3.6.1 Protection from External Hazards

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

3.6.1.1 Seismic

The licensee's storage philosophy for the FLEX equipment is to store two sets of equipment, one set in each of two storage buildings. One storage location is a newly constructed building called the "FLEX Building." The other storage location is an existing structure, modified for FLEX storage, called "Warehouse 6." The strategy of using two separate buildings is employed primarily to account for tornado missiles via building separation. The licensee originally evaluated both storage locations to an earthquake loading that was lower than the accelerations associated with the site SSE. The NEI 12-06 definition of "robust" states that to be considered robust, structures must either meet the plant's current design basis for the applicable hazard or be shown by analysis or test to meet or exceed the current design basis. During the audit process, the NRC staff questioned the ability to deploy the FLEX equipment if an earthquake at the magnitude of the SSE were to occur. By letter dated February 20, 2017 [Reference 18], the licensee responded to this question. The licensee stated that the FLEX Building was reevaluated to show that the wind loadings used in the building design bound the loadings from an SSE-level earthquake. Based on the results of this analysis, the staff considers the FLEX Building to be seismically robust. With a full set of equipment deployable from the FLEX Building, the licensee has the ability to deploy "N" sets after a seismic event, thus meeting NEI 12-06, Section 1.1. The licensee did not evaluate Warehouse 6 for seismic capability up to the SSE level. Therefore, based on the NEI 12-06 definition of "robust", the NRC staff does not consider Warehouse 6 to be seismically robust. Thus, when equipment stored in the FLEX Building is out-of-service, such as for periodic maintenance, the licensee would not have "N" sets protected from a seismic event. In this situation, the provisions of NEI 12-06, Section 11.5.4.b would limit the allowed out-of-service time for the unavailable equipment in the FLEX Building to 45 days, versus 90 days. This reduction could be alleviated by either moving "N" sets of equipment to a protected location, or by showing by analysis or test that Warehouse 6 is robust for seismic considerations (capable of withstanding the SSE).

3.6.1.2 Flooding

Neither of the licensee's two designated storage locations for the FLEX equipment are protected against the design-basis flood. The FLEX Building is located at a lower site elevation than Warehouse 6, but both would be impacted by the design-basis flood. For a flood predicted to exceed the site grade of 930 feet, the licensee's strategy is to use the flood warning period to build a horseshoe-shaped levee around the critical plant structures and move the FLEX equipment from both buildings inside of this berm-protected area before the flood waters arrive at a level that would inhibit deployment. This strategy would correspond to the provisions of NEI 12-06, Section 6.2.3.1.c. A design-basis flood would also necessitate the construction of a temporary site access road to support the coping strategy while the normal site access road is flooded.

During the audit process the NRC staff reviewed the applicable Monticello procedures to confirm that the licensee's administrative controls reflect this strategy. The staff reviewed the licensee's procedure A.6, "Acts of Nature," Revision 56, with respect to external flooding. This procedure provides an overall operational coordination of external event preparations and response. The staff noted that procedure A.6 invokes procedure 8300-02, "External Flooding Protection Implementation to Support A.6, Acts of Nature". Parts "O" and "P" of procedure 8300-02 direct the construction of the levee and temporary access road when river level projections exceed 930 feet. Movement of FLEX equipment is directed by procedure C.5-4103, "FLEX Response during External Flood," Revision 0, also invoked from procedure A.6. Procedure C.5-4103 indicates that movement of equipment out of the FLEX Building should occur when projected river level exceeds 919 feet and equipment is moved out of Warehouse 6 when projected level exceeds 930 feet. According to the licensee's FIP, the site expects to request NSRC equipment before the site access road is inundated. Depending on the timing of the normal access road becoming flooded, delivery of NSRC equipment, as well as any subsequent deliveries, could be accommodated by the temporary access road.

3.6.1.3 High Winds

For high wind (tornado) conditions the licensee's strategy is to utilize building separation with two sets of equipment stored in diverse locations to ensure that one "N" set survives the initiating event, in accordance with NEI 12-06, consideration 7.3.1.1.c. The licensee's two storage locations (FLEX Building and Warehouse 6) are separated by over 900 feet. The licensee evaluated the building separation and the predominant path of tornados in choosing a location for construction of the FLEX Building and utilization of Warehouse 6. After a postulated high wind event, a deployment decision would be made during the site assessment. Debris removal equipment is stored in each building to facilitate deployment.

During the audit process the NRC staff confirmed that the licensee procedure C.5-4101, "FLEX Site Assessment," Revision 1, makes an assessment of the staging areas and FLEX storage locations to determine a deployment plan, given the site conditions. The staff also confirmed that the licensee's calculation, 14-010 "MNGP FLEX Storage Location Separation Report," Revision 0, calculated a minimum 850 foot separation distance based on data specific to the site location, and thus the licensee's building separation is acceptable.

3.6.1.4 Snow, Ice, Extreme Cold and Extreme Heat

According to the licensee's FIP, FLEX equipment is stored in buildings that maintain a minimum temperature of 40°F. This ensures that equipment will start when called upon to function. Snow removal is a normal activity because of the typical winter conditions present at the plant site. Reasonable access to FLEX equipment is maintained prior to and throughout a snow event. Ice management is performed, as required, such that large FLEX equipment can be moved by vehicles. Debris removal equipment will move through any snow accumulations and can also be used to move portable equipment, if needed. Snow removal is addressed by plant procedure. During the audit process, the NRC staff reviewed the licensee's program document, "Diverse and Flexible Coping Strategies (FLEX) Program Document," Revision 2, and noted that it recognizes that snow, ice and extreme cold hazards can present a challenge to FLEX deployment because the extreme conditions postulated to cause the ELAP could remain in place during deployment of FLEX equipment. This is consistent with the discussion contained in NEI 12-06, Section 8.

The two FLEX storage locations are designed to American Society of Civil Engineers (ASCE) standard ASCE 7-10, with respect to snow, ice, and extreme cold considerations. According to the licensee's program plan, the heating provisions of the storage locations will provide adequate heating to prevent equipment from freezing, and are designed to withstand required snow and ice loads. The licensee expects to deploy equipment from the storage locations within 12 hours of ELAP initiation, and thus the loss of building heating for the period of time after the ELAP initiation will not inhibit deployment.

The licensee's FIP states that high temperature does not impact the deployment of FLEX equipment because high temperature does not create obstacles for deployment paths. All FLEX equipment was procured to be suitable for use in peak temperatures for the region.

3.6.1.5 Conclusions

Based on this description, 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 Availability of FLEX Equipment

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

The licensee's strategy generally incorporates the "N+1" provisions contained in NEI 12-06 by storing two sets of equipment in two storage buildings, spatially separated for tornado considerations. As discussed in Section 3.6.1.1 of this safety evaluation, only one of the storage buildings has been shown to be robust for seismic events. Thus, the licensee's management of allowed out-of-service time must carefully consider seismic protection when equipment stored in the seismically robust FLEX Building is unavailable.

After the issuance of NEI 12-06, Revision 0, NEI, on behalf of the commercial nuclear power industry, submitted a letter to the NRC [Reference 52] proposing an alternative regarding the quantity of spare hoses and cables to be stored on site. The alternative proposed was that either: (a) 10 percent additional lengths of each type and size of hoses and cabling necessary for the "N" capability plus at least one spare of the longest single section/length of hose and cable be provided, or (b) that spare cabling and hose of sufficient length and sizing to replace the single longest run needed to support any FLEX strategy. By letter dated May 18, 2015

[Reference 53], the NRC agreed that the alternative approach is reasonable, but that the licensees may need to provide additional justification regarding the acceptability of various cable and hose lengths with respect to voltage drops, and fluid flow resistance. The NEI alternative for spare hoses and cables was later incorporated into the NRC-endorsed NEI 12-06, Revision 2. The NEI alternative postulates that the most probable cause for degradation/damage to hoses and cables would occur during deployment of the equipment.

The licensee has elected to store the majority of their FLEX equipment in two separate storage buildings. For non-hurricane sites such as Monticello, this concept provides protection from high wind missiles via separation, in that the equipment in at least one building should remain deployable because a tornado is not postulated to impact both structures. Thus, in order to ensure that sufficient spare hoses and cables are available, after accounting for the potential of damage during deployment, there would need to be appropriate spares in each storage building.

In its FIP, Section 11.6, the licensee described its strategy for hoses and cables. The licensee states that full sets are stored in each FLEX storage location along with additional hoses and cables. During the audit process, the staff reviewed the licensee's provisions for spare hoses and cables in more detail, specifically corrective action program item 501000001110, "Additional 480V Cables Required for FLEX," and procedure OSP-FIR-1489, "B.5.B/FLEX Equipment Inventory," Revision 21. These documents confirm that the quantities of spare hoses and cables in each FLEX storage location include the appropriate spare quantity.

Since the licensee's FIP description of spare hoses and cables, as clarified by the site's implementing documents, appears to be consistent with the provisions of NEI 12-06, Revision 2, and the FIP states that the licensee is following NEI 12-06, Revision 2, the NRC staff concludes that the quantity of spare hoses and cables for FLEX at Monticello is acceptable.

Based on the number of portable FLEX pumps, FLEX DGs, and support equipment identified in the FIP and during the audit review, the NRC staff finds that, if implemented appropriately, the licensee's FLEX strategies include a sufficient number of portable FLEX pumps, FLEX DGs, and equipment for RPV makeup and core cooling, SFP makeup, and maintaining containment consistent with the "N+1" provision in Section 3.2.2.16 of NEI 12-06.

3.7 Planned Deployment of FLEX Equipment

3.7.1 Means of Deployment

According to the licensee's FIP, the Phase 2 FLEX strategies rely upon two trucks, a front end loader, and a forklift to accomplish deployment. This equipment will be used to: (a) transport FLEX equipment and site personnel, and (b) clear the deployment routes, staging areas, and paths for running hoses and cable. The two trucks are the primary tow vehicles for deployment, however the front end loader and forklift are capable of performing towing if not needed for debris removal. For Phase 3 the NSRC-equipment will be delivered by the SAFER group to Staging Area "B" and deployment and equipment check-out will be coordinated between SAFER and the site organization.

3.7.2 Deployment Strategies

According to the FIP, the Monticello site is not susceptible to liquefaction that could otherwise inhibit deployment after a seismic event. During the audit process, the staff reviewed USAR

Section 2.5.5, to confirm that the site is not susceptible to liquefaction based on the soil properties and design-basis earthquake accelerations. Site borings used in this evaluation as shown in USAR Figure 2.5-2 cover the expected deployment paths. For a flooding event the licensee's deployment strategy is different and the licensee expects to deploy equipment prior to flood water impacting the deployment routes. For high wind events, deployment of FLEX equipment could be impaired by large debris. Debris removal equipment is stored with the FLEX equipment to ensure the ability to provide a clear path for deployment of FLEX equipment is available. The licensee's strategy assumes that heavy debris removal, such as what would be accomplished by operating the front end loader and forklift, would be accomplished by supplemental personnel arriving at the site after 6 hours into the event. Thus, on-shift personnel initially will perform light debris removal and perform hose/cable layout. During the audit process, the staff reviewed procedure C.5-4101, "FLEX Site Assessment," Revision 1, and confirmed that it evaluates site conditions, available FLEX equipment, available water sources, and establishes preferred staging locations such that FLEX deployment can be optimized. The staff also confirmed during the audit process that the licensee's staffing analysis and FIP Table 10.0-1 (event timeline) are consistent with heavy debris removal not beginning for at least 6 hours into the event.

According to the FIP, snow removal is a normal activity at the plant site because of the climate and the licensee stated that it has established provisions for the stored FLEX equipment such that access is maintained prior to and throughout a snow event. The NRC staff confirmed that access to stored FLEX equipment is maintained prior to and throughout a snow event by reviewing Action Request 01455074 during the audit process. According to the FIP, ice management is also a normal plant activity and would be performed such that the large FLEX equipment can be moved by the vehicles provided in the program. Ice melting chemicals and equipment are maintained on site to support this function. The licensee's FIP states that debris removal equipment (the front end loader and/or large forklift) was procured to be able to move through snow accumulations as well as to be used to move portable equipment, if needed.

The FLEX strategy uses the discharge canal as an option for the makeup water source in Phases 2 and 3. According to the licensee, this canal is unlikely to have any significant ice buildup because it is the discharge path for normal plant cooling operations. Therefore, especially for the majority of the time when the plant had been at power prior to the event, warm water would be present in the discharge canal prior to the onset of loss of offsite of power such that makeup water would not be inhibited by ice during the response to an ELAP with loss of normal access to the UHS event.

3.7.3 Connection Points

3.7.3.1 Mechanical Connection Points

Core Cooling

Sections 3.1.5.1 and 3.1.5.2 of the licensee's FIP 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 RHRSW system crosstie to the RHR system. Figure 3.1.2-1 of the licensee's FIP shows this connection being made in the "A" RHRSW train. This connection is in the Turbine Building. The alternate connection is made to the "A" RHR train, in the RHR Division I room, located in the Reactor Building. Both the Reactor and Turbine Buildings are Class I structures designed to be robust with respect to earthquakes, wind loading, tornado loading, and tornado generated missiles. The licensee also has the capability to make an additional connection to the fire system located near the site cooling towers. This connection is not robust for all external events but could be utilized if it is available.

SFP Cooling

Sections 3.2.4.1 and 3.2.4.2 of the licensee's FIP describe the primary and alternate SFP connections. Similar to core cooling, the portable diesel-driven FLEX pump is also used for SFP cooling. The FLEX pump's primary connection to the SFP is through the RHRSW connection described in the core cooling section. From there, water can be directed to either the fire system or the SFP cooling system. For the alternate strategy, hose is run from the discharge of the FLEX pump directly to the SFP. No physical connections to permanent plant equipment are required for the alternate strategy.

3.7.3.2 Electrical Connection Points

During Phase 2, the licensee's strategy is to supply the necessary electrical power using a combination of permanently installed and portable components. The main power supply for this portion of the strategy is a 200 kW, 480 Vac FLEX DG. The preferred deployment area of this FLEX DG is near the south side of the Plant Access Building (PAB). The alternate deployment area is east of the PAB. The licensee's primary strategy uses installed inlet receptacles near each Class 1E battery charger. Each receptacle is connected via conduit and cable to a breaker installed in each battery charger to receive power from the FLEX DG. A mechanical interlock was added to each battery charger that allows either the normal ac input breaker to be closed or the input breaker from the 480 Vac receptacle to be closed, but not both. With only one input breaker closed, the 480 Vac receptacle is isolated from the normal ac input to the battery charger. The 480 Vac inlet receptacle circuits will provide the capability to repower the 125 Vdc/250 Vdc battery chargers from a 200 kW FLEX DG. The receptacles, cabling and connections are installed to protect against seismic events and are located in a seismically robust structure which also provides protection from missiles, flood, snow and ice. During the audit process the staff reviewed the licensee's procedure C.5-4402, "Stage and Connect FLEX 480V Portable Diesel Generator," Revision 0, to confirm that it provides adequate direction for connecting the 200 kW FLEX DG.

The licensee's alternate strategy is to utilize 480 Vac connections located at the motor control centers (MCCs) that feed the battery chargers. Modified spare breaker connections were fabricated to connect the MCCs to the 200 kW FLEX DG. Each receptacle uses a spare breaker connection in the MCC to backfeed the associated battery charger. Each connection will receive power from the 200 kW FLEX DG. The MCC spare breaker circuits will provide the capability to re-power the 125 Vdc/250 Vdc battery chargers from the 200 kW FLEX DG. According to the licensee's FIP, the MCCs are installed to protect against seismic events and are located in a seismically robust structure which also provides protection from missiles, flood, snow and ice. During the audit process the staff reviewed the licensee's procedure C.5-4405, "Backfeed MCCs from 480V Portable Diesel Generator," Revision 0, to confirm that it provides direction for connecting the 200 kW FLEX DG and backfeeding the MCCs.

For Phase 3, the licensee will receive two 1 MW 4160 Vac CTGs and one 1100 kW 480 Vac CTG from an NSRC. The NSRC supplied 4160 Vac CTGs would be deployed near the Turbine Building railroad doors or west of the Turbine Building addition. Power would be restored to one of two 4160 Vac Class 1E switchgear cubicles. The 4160 Vac CTGs will be used to power one division of essential power to support continued core cooling functions. During the audit process the staff reviewed licensee procedure C.5-4410, "FLEX 4kV Generator," Revision 2, which provides direction for staging and connecting the 4160 Vac CTGs. The staff confirmed that procedure C.5-4410 provides guidance and direction to verify proper phase rotation (via a work order) when connecting the 4160 Vac CTGs.

3.7.4 Accessibility and Lighting

According to the licensee's FIP, plant lighting is supplied by wall-mounted battery-powered lighting units in the control room and other critical areas. Since the installed emergency dc lighting is load shed in the licensee's strategy, use of the wall-mounted units along with portable flashlights is credited for use in Phase 1 of the event. During the audit process, the NRC staff reviewed modification number 86M023, Revision 1, to confirm that the wall-mounted units are seismically robust. In Phase 2, after the FLEX DG is operating, control room emergency lighting will be available because the 125 Vdc system will be restored. In addition, according to the licensee's FIP, portable lights are available for use in the deployment of FLEX equipment such as the yard areas. These lights will either be battery powered, or will be powered by the 120 Vac FLEX DGs. The licensee's FIP indicates in Table 11.5-1 that 18 lighting units are stored in the two FLEX storage locations. Thus, they are stored in a protected location so that they would be available for deployment. During the audit process, the NRC staff observed that the licensee has developed an FSG to help control deployment of the supplemental lighting.

3.7.5 Access to Protected and Vital Areas

By letter dated October 12, 2015 [Reference 19], the licensee stated that keys are available to allow operators to access internal locked areas in the plant during an ELAP. During the audit process, the NRC staff confirmed that the licensee has administrative controls in place to ensure that access to areas required for the ELAP response is feasible, assuming that the normal access control systems are without power.

3.7.6 Fueling of FLEX Equipment

Section 5.3.4 of the licensee's FIP states that fuel can be obtained from the EDG fuel oil day tanks. Each tank has a 1,500 gallon capacity and is located in a seismic Class I structure. The licensee will deploy two FLEX tow vehicles that have fuel storage cubes mounted on the truck bed with a capacity of 264 gallons each to supply fuel for the diesel-driven portable equipment. Each truck also has a transfer pump used to transfer fuel to and from the fuel cubes. Based on the design and location of these EDG day tanks, the staff concludes that the day tanks are robust and the fuel oil contents should be available to support the licensee's FLEX strategies during an ELAP event.

As stated above, the EDG fuel oil day tanks have approximately 3,000 gallons total between the two tanks. The licensee stated in the FIP that the 72-hour total consumption for all FLEX equipment is 1,800 gallons. Furthermore, the licensee has calculated that operators will have to begin refueling the first components at 22.2 hours. Given the fuel demand for the Phase 2 FLEX components cited above and almost double the amount of available fuel, the staff concludes that the licensee has a sufficient inventory of fuel for diesel-powered equipment required for the FLEX strategy until additional fuel arrives from off-site. Furthermore, the staff finds that the licensee should be able to refuel the diesel-powered equipment used in the FLEX strategy such that uninterrupted operation is ensured.

Section 5.3.4 of the licensee's FIP states that all FLEX equipment will be stored in a fueled condition. It also states that fuel oil in the fuel tanks of the portable diesel engine driven FLEX equipment is maintained in accordance with the preventative maintenance (PM) program. The FIP also states that the EDG fuel oil quality is maintained under technical specification controls. Based on the FIP description, the staff concludes that the licensee has demonstrated that fuel oil quality will be maintained to support the overall FLEX strategy.

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 Monticello 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.

By letter dated September 26, 2014 [Reference 27], 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.

During the audit process, the NRC staff reviewed AREVA Document Number 38-9233760-000, "SAFER Response Plan for Monticello Nuclear Generating Plant," Revision 1, dated January 23, 2015. The staff noted that the 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. The staff also noted that the plan contains provisions for helicopter support, if it is required.

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 25-35 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 Monticello, the SAFER plan does not have provisions for Staging Area "D". Staging Area "C" is the Maple Grove Service Center located approximately 26 miles from the site. The Maple Grove facility is owned by Xcel Energy. Staging Area "B" is a spare lot within the Xcel Energy owned property surrounding the Monticello site, behind the contractor parking lot, across from the site receiving warehouse. Staging Area "A" corresponds to the various deployment locations for each piece of FLEX equipment on the site.
Use of helicopters to transport equipment from Staging Area "C" to Staging Area "B" is described within the Monticello SAFER Plan. For helicopter operations, the licensee plans to use the Elm Creek Park Reserve, approximately 3 miles from the Maple Grove facility, as a staging area for helicopter operations due to the highly populated area around the Maple Grove facility. If this option were needed, equipment would be trucked to the Elm Creek Park Preserve and airlifted from there to the site Staging Area "B". The staff review of Monticello's plan concludes that the helicopter provisions are appropriate.

For a flooding event deployment Staging Area "B" may not be available. In this case the licensee will stage the NSRC equipment within the area protected by the horseshoe berm, southwest of the power block.

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 Monticello, 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 which occurs with the loss of forced ventilation in areas that continue to have heat loads. According to the licensee's FIP, loss of ventilation analyses were performed to quantify the maximum steady-state temperatures expected in specific areas related to their FLEX mitigation strategy implementation to ensure that the environmental conditions remain acceptable and within equipment qualification limits. The key areas identified for all phases of execution of the licensee's FLEX strategy activities are the control room, PAB (battery rooms including chargers), Emergency Filtration Train (EFT) Building (battery room, battery charger room, and inverter room), RCIC pump room, and containment. The FIP concludes that adequate ventilation is available to support operating FLEX equipment if high temperatures occur inside plant buildings. Using the audit process, the NRC staff confirmed the licensee's FIP statements by reviewing calculations and procedures for the following specific areas of concern during the postulated event.

Control Room

Licensee calculations 90-038, "Control Room Space Temperature Evaluation During Station Blackout," Revision 4 and 91-056, "Control Room Electrical Heat Losses for SBO and ELAP, Revision 1, modeled the transient temperature response in the control room following an ELAP event. The calculations established a maximum control room temperature of 120°F and determined the maximum electrical heat load to maintain control room temperature below this temperature. The calculations showed that the temperature in the control room was maintained below 120°F. The staff confirmed that the licensee's procedure C.5-4502, "Control Room and PAB Ventilation During FLEX Conditions," Revision 0, provides guidance to block open doors in the control room and PAB, and establish forced ventilation consistent with the calculation assumptions.

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

Class 1E Battery Rooms (PAB)

Licensee calculation 15-017, "PAB/EFT Battery Rooms Heat-up During a SBO and ELAP," Revision 0, modeled the transient temperature response in the battery rooms located in the PAB following an ELAP event. The analysis showed that the maximum temperature in battery room 167A (D1), 167B (D3), and 167 (D2) stabilized at 102.7°F, 107.5°F, and 103.5°F, respectively. The staff review noted that procedure C.5-4502, provides guidance to block open doors and establish forced ventilation.

Based on the above, the NRC staff finds that the licensee's ventilation strategy (opening doors and establishing forced ventilation) should maintain battery room temperature below the maximum temperature limit (122°F) of the batteries, as specified by the battery manufacturer (C&D Technologies). Therefore, the batteries should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Class 1E Battery Room (EFT Building)

Licensee calculation 15-017, "PAB/EFT Battery Rooms Heat-up During a SBO and ELAP," Revision 0, modeled the transient temperature response in the battery room located in the EFT Building following an ELAP event. The analysis showed that the maximum temperature in the battery room stabilized at 90.2°F. The staff's review noted that procedure C.5-4503, "EFT Ventilation During FLEX Conditions," Revision 1, provides guidance to block open doors and establish forced ventilation in the battery room, battery charger room, and the inverter rooms.

Based on the above, the NRC staff finds that the licensee's ventilation strategy (opening doors and establishing forced ventilation) should maintain battery room temperature below the maximum temperature limit (122°F) of the batteries, as specified by the battery manufacturer (C&D Technologies). Therefore, the batteries should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

RCIC Pump Room

Licensee calculation 14-112, "Reactor Building Heat-up During an Extended Loss of AC Power (ELAP)," Revision 0, modeled the transient temperature response in the RCIC pump room following an ELAP event. The calculation showed that the maximum temperature in the RCIC pump room is 125.5°F. This temperature is based on the licensee opening doors within 2 hours and establish forced ventilation utilizing a fan with a minimum of 1,400 cubic feet per minute (cfm) air flow within 12 hours of an ELAP event. The staff's audit review confirmed that procedure C.4-B.09.02.A, "Station Blackout," Revision 46* provides guidance to open doors within the first 30 minutes of an ELAP event. Procedure C.5-4501, "Reactor Building Ventilation During FLEX Conditions," Revision 0, provide guidance to establish forced ventilation in the RCIC pump room.
Based on temperatures remaining below the equipment environmental limits (150°F in accordance with USAR Section 10.2.5.3.1), the NRC staff finds that the electrical equipment in the RCIC Pump Room will not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Containment

Licensee calculation PRA-CALC-14-003," Monticello MAAP Thermal Hydraulic Calculations to Support Extended Loss of AC Power (ELAP) Mitigating Strategies," Revision 1, modeled the transient temperature response in the containment following the first 48 hours of ELAP event. The containment design pressure and temperature limits (62 psig and 281°F) bound the expected containment pressure and temperature values (30.7 psig and 240°F) calculated for an ELAP. The results of the licensee's analysis indicated that the peak temperature in the drywell would be approximately 250°F and would stabilize at 240°F. Consistent with the analysis, the licensee plans to vent the torus via the HPV system to maintain containment parameters below design limits.

During the audit process the staff reviewed licensee environmental qualification (EQ) report 98-004, "ASCO Solenoid Valves (50.49)", Revision 1, which showed that the SRV solenoid valves located in the containment are tested and qualified to a maximum temperature of 346°F. The components are expected to function for 180 days post-accident. Licensee EQ report 98-075, "Weed Thermocouples (50.49)," Revision 1, showed that the temperature elements located in containment are qualified to a temperature of 500°F. Both reports show that there is significant margin remaining between equipment qualification test data and impact of post ELAP temperatures.

The licensee will receive offsite resources and equipment from an NSRC following 72 hours after the onset of an ELAP event at Monticello. The NRC staff finds that it is reasonable to expect that the licensee could utilize these resources to reduce or maintain temperatures within containment to ensure that required electrical equipment survives indefinitely, if necessary. Nonetheless, plant operators will continue to monitor containment parameters and perform additional actions that may be required to reduce containment temperature and pressure.

Based on temperatures remaining below the design limits of equipment, the ability to vent he torus via the HPV system, and the availability of offsite resources after 72 hours, the NRC staff finds that the electrical equipment in the containment 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 control room, battery rooms, RCIC pump room, and containment, the NRC staff finds that the electrical equipment should perform their required functions at the expected temperatures as a result of a loss of ventilation during the postulated event.

3.9.1.2 Loss of Heating

The licensee's FIP, Section 7.2, states that no heat traced components are relied upon for using the FLEX strategies. In Section 5.2.1.4 of its FIP, the licensee also states that bypass flow paths are available for continuous flow to prevent hoses from freezing. It also states that the FLEX equipment is stored in buildings that are maintained above 40°F. Additionally, the licensee states heaters and canvas materials will be available onsite to keep FLEX equipment running during extreme cold conditions.

The Monticello Class 1E station battery rooms located inside the PAB and EFT Building. At the onset of the event, they 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 rooms are not expected to be sensitive to extreme cold conditions due to their location in the PAB and EFT 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, according to the licensee's FIP, Section 7.1, HPV components (HPV battery) on the third floor of the EFT Building could be susceptible to extreme low temperatures during an ELAP event. Also according to the FIP, operators are instructed to add portable heaters as needed within 15 hours upon initiation of an ELAP to maintain EFT Building third floor temperatures above 40°F. During the audit process, the staff reviewed calculation 16-055, "Monticello Gothic Analysis for the Hardened Containment Vent Project," Revision 1, which analyzed the temperature transient during an ELAP for third floor of the EFT Building and determined that heating is required to maintain room temperature above 40°F. The staff also confirmed that procedure C.5-4503 provides guidance to establish temporary heating on the third floor of the EFT Building to address this concern.

Based on the licensee's FIP, the NRC staff finds that Monticello FLEX equipment, Class 1E station batteries, and HPV batteries should perform their required functions as a result of loss of normal heating during an ELAP event.

3.9.1.3 Hydrogen Gas Control in Vital Battery Rooms

An additional ventilation concern that is applicable to Phases 2 and 3 is the potential buildup of hydrogen in the battery rooms as a result of loss of ventilation during an ELAP event. Off-gassing of hydrogen from batteries is only a concern when the batteries are charging. According to the licensee's FIP Section 7.1 ventilation is provided to mitigate the potential for hydrogen buildup in the applicable locations. The staff confirmed that licensee procedures C.5-4502 and C.5-4503 direct the operators to establish portable ventilation in the PAB and the EFT Building. The ventilation fans would be powered by portable FLEX 12 kW DGs.

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

3.9.2 Personnel Habitability

According to the licensee's FIP, analyses were performed to quantify the maximum steady state temperatures to ensure that environmental conditions remain acceptable for personnel habitability and to ensure that access will be available as long as necessary. The FIP also provided a listing of key areas for operator access and any necessary procedure actions to address habitability considerations.

The licensee's actions to ensure habitability primarily occur in Phase 2 of the event where supplemental ventilation would be provided, as necessary, for the operators. Phase 2 ventilation strategies will remain in effect until off-site resources provided by the NSRC arrive on site and the ERO can review the availability of plant equipment along with off-site resources to develop plans to restore plant systems.
In order to confirm the licensee's FIP assertion that environmental conditions remain acceptable for personnel habitability, the staff reviewed various licensee calculations and procedures.

3.9.2.1 Main Control Room

During the audit process, the staff reviewed calculation 90-035, "Control Room Temperature during a Station Blackout," Revision 4. The calculation used the GOTHIC computer program, established a maximum room temperature, and determined the maximum electrical heat load to maintain the room at or below the maximum temperature established in the calculation. The calculation assumed an initial control room temperature of 80°F and a constant outdoor ambient temperature of 104°F. The calculation assumed the control room to be occupied by 10 people. The calculation did not assume any doors were opened to establish ventilation flow paths until after 24 hours nor did it assume any ventilation provided by portable fans. The calculation determined that to maintain the control room at or below 120°F, the maximum electrical load is 20,000 watts for the first 2 hours followed by a load shed to 9600 watts for the remainder of the event.

Guidance document NEI 12-06, Revision 2, indicates that the effects of a loss of ventilation in an ELAP can be addressed consistent with NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors". NUMARC 87-00, 2.7.2, "Effects of Loss of Ventilation, Basis" indicates that during a 4 hour coping time for a SBO a bulk air temperature of 120°F would not adversely affect operability of most electrical and mechanical equipment and instrumentation. It also indicates that during the 4 hour coping time a bulk air temperature of 110°F is tolerable for personnel performing light work such as control room operators.

During the audit process, NRC staff inquired what the actual expected electrical loads will be during an ELAP in order to verify that the calculation enveloped the actual conditions expected. The licensee provided calculation 91-056, "Control Room Electrical Heat Losses," Revision 1, during the audit process to show the control room electrical heat contribution during an ELAP. The analysis concludes that the heat load for the first 2 hours is 17,958 watts, and the heat load after 2 hours is 4,757 watts. Therefore, based on the audit review, the staff concludes that calculation 90-035, Revision 4, envelopes the expected electrical loads.

During the audit process, the NRC staff also requested an evaluation of personnel habitability since the licensee's evaluation of the 120°F control room temperature was based on equipment performance. The licensee responded by performing engineering evaluation number 608000000035, "Control Room Space Temperature Evaluation," dated September 27, 2017.

This evaluation noted that several assumptions in calculation 90-035 are conservative. For example, it uses a continuous outdoor temperature of 104°F with no diurnal variation. No doors are assumed to be opened nor any portable fans used to provide ventilation. In addition, the actual electrical heat loads after 2 hours are almost half of the calculated allowable for a 120°F control room temperature. The evaluation further calculated that with an approximate 5000 watt electrical heat load, assuming a more realistic ambient outdoor temperature of 95°F (2013 ASHRAE [American Society of Heating, Refrigerating and Air-Conditioning Engineers] Fundamentals Handbook, maximum outdoor air temperature for St. Cloud, Minnesota) and providing a minimum air flow of approximately 1,035 cfm will maintain the control room temperature below 110°F. The 1,035 cfm assumption is conservative with respect to the portable FLEX fan nameplate rating of 4,650 cfm. Thus, the licensee concluded that the postulated control room environment would not adversely impact the operators during an ELAP event. The NRC staff agreed with the conclusions of the licensee's evaluation and confirmed that procedure C.5-4502 provides guidance for opening selected doors and using portable fans and blowers to provide ventilation/cooling, consistent with the engineering evaluation.

3.9.2.2 Spent Fuel Pool Area

As the SFP temperature increases, the additional moisture and heat from the SFP will enter the local atmosphere in the Reactor Building. The staff noted that the licensee's sequence of events timeline in the FIP indicates that operators will initiate makeup to the SFP prior to 57 hours into the event to ensure the SFP area remains habitable for personnel entry.

During the audit process, the staff confirmed that procedure C.4-B.09.02.A, "Abnormal Procedures, Station Blackout", Revision 46, directs operators to establish a natural convection air flow. Setup of SFP makeup activities is initiated in C.4-B.09.02.A, after natural ventilation pathways have been established. Procedure C.5-4301, "Spent Fuel Pool Makeup with FLEX Portable Diesel Pump", Revision 2, directs operators to complete setup activities prior to the occurrence of boiling in the SFP because of habitability concerns.

3.9.2.3 Other Plant Areas

Based on the temperature evaluations described in Section 3.9.1 of this safety evaluation, and the limited time of access that would be necessary for those areas, the staff concludes that there should not be any habitability issues for other plant areas.

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

Condition 3 of NEI 12-06, Section 3.2.2.5 states that cooling and makeup water inventories are considered available if they are contained in systems or structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles. The NRC staff reviewed the licensee's planned water sources to verify that each water source was robust as defined in NEI 12-06.

3.10.1 RPV Makeup

As described in the FIP, the non-seismically qualified CST is the normal suction for the RCIC pump. The FIP states that the RCIC suction will automatically swap to the suppression pool in event that the CST is unavailable. Based on the licensee's FIP timeline, the suppression pool will be the RCIC suction source for approximately 10-11 hours. The suppression pool is located in the Reactor Building and is listed as a Class I structure that is part of the primary containment, so it is protected from all applicable hazards. Based on the FIP description, the NRC staff concludes that the suppression pool is robust.

After the 10-11 hour timeframe, the portable FLEX pump would be aligned to provide RPV makeup and core cooling. The water source for the FLEX pump is the river with a suction taken from either the intake or the discharge canal. However, other, non-robust, sources of water that may be available and used prior to the discharge canal as they provide cleaner water. In the FIP, Section 2.3.4.8 states that the intake and discharge canals will be available during all external hazards. Based on the FIP description of the river water sources, the NRC staff concludes that the intake and discharge canal sources are robust.

3.10.2 Suppression Pool Makeup

Based on a review of the FIP, the licensee's plan does not contain provisions for providing makeup to the suppression pool to support the Monticello FLEX response strategy.

3.10.3 SFP Makeup

No SFP makeup is required in Phase 1. In Phase 2, makeup to the spent fuel pool is from the intake or discharge canal via the portable FLEX pump. The water sources for the FLEX pump are the same as those described above for RPV makeup and are therefore robust.

3.10.4 Containment Cooling

The licensee's plan does not provide any water sources used to support a containment cooling function.

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, at least 71 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 stated that they have the ability to implement makeup to the SFP within that time. During the audit process the staff confirmed that the licensee's procedures would implement SFP makeup well before reaching a level that would uncover fuel assemblies.

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 of NEI 12-06 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 NSPM has adopted the BWROG paper TP-15-019, "BWR-Specific Shutdown Refueling Mode Guidance," Revision 0, as the basis for the Monticello FLEX strategy in shutdown and refueling modes. According to the licensee, the BWROG paper expands upon the risk management concepts described in the NRC-endorsed in NEI position paper on shutdown and refueling mode conditions that have been incorporated into NEI 12-06, Revision 2, Section 3.2.3. Specifically, the BWROG paper defines certain shutdown/refueling conditions, and identifies actions that should be considered as part of the outage risk management program that are commensurate with the risk and applicable for a given plant state.

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

According the licensee's FIP, the overall plant response to an ELAP will be accomplished through normal plant command and control procedures and practices. The FSGs are utilized to provide direction for using FLEX equipment in maintaining or restoring key safety functions. When FLEX equipment is needed to supplement EOP or Abnormal Operating Procedure (AOP) strategies, the EOP or AOP directs the entry into and exit from the appropriate FSG.

According to the licensee, the FSGs have been reviewed and validated by the involved groups to the extent necessary to ensure that implementation of the associated FLEX strategy is feasible. In addition, FSG maintenance is performed by the station procedures group.

3.12.2 Training

According to the licensee's FIP, Monticello's 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 BDBEE 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.

3.12.3 Conclusions

Based on the description above, the NRC staff concludes that the licensee's procedure development, issuance, and control are in accordance with NEI 12-06, Section 11.4. In addition, based on the description above, the staff concludes that a training program for BDBEE mitigation has been established and should be maintained in accordance with NEI 12-06, Section 11.6.

3.13 Maintenance and Testing of FLEX Equipment

As a generic issue, NEI submitted a letter to the NRC dated October 3, 2013 [Reference 43], which included EPRI Technical Report 3002000623, "Nuclear Maintenance Applications Center: Preventive Maintenance Basis for FLEX Equipment." By letter dated October 7, 2013 [Reference 44], the NRC endorsed the use of the EPRI report and the EPRI database as providing a useful input for licensees to use in developing their maintenance and testing programs. In its FIP, the licensee stated that they would conduct maintenance and testing of the FLEX equipment in accordance with the industry letter.

In its FIP, the licensee described that initial component level testing, consisting of factory acceptance testing and site acceptance testing, was conducted to ensure the portable FLEX equipment can perform its required FLEX strategy design functions. Factory acceptance testing verified that the portable equipment performance conformed to the manufacturers rating for the equipment as specified in the purchase order. Verification of the vendor test documentation was performed as part of the receipt inspection process for each of the affected pieces of equipment and included in the applicable vendor technical manuals. Site acceptance testing confirmed portable equipment delivered to the site performed in accordance with the functional design requirements.

According to the licensee, the portable BDBEE equipment that directly performs a FLEX mitigation strategy is subject to periodic maintenance and testing in accordance with the provisions of NEI 12-06 and Institute of Nuclear Power Operations document AP-913, "Equipment Reliability Process Description," to verify proper function. Maintenance and testing of FLEX equipment is governed by the NSPM PM program. The NSPM PM program utilizes the EPRI Preventive Maintenance Basis Database as an input to the development of fleet specific Monticello PM basis templates. According to the licensee, manufacturer's recommendations were used when templates were not available from EPRI.

The PM templates include activities such as:

- Periodic static inspections
- Fluid analysis
- Operational inspections

- Periodic functional verifications
- Periodic performance verification tests

Additional FLEX support equipment and vehicles that require maintenance and testing will have PMs and testing controlled by the Monticello work order process to ensure they will be able to perform their required functions during a BDBEE.

According to the licensee's FIP, unavailability controls for FLEX equipment conforms to the guidance of NEI 12-06, Revision 2, as follows:

- Portable FLEX equipment may be unavailable for up to 90 days provided that the site FLEX capability ("N" set) is available.
- Connections to permanent equipment required for FLEX strategies can be unavailable for 90 days provided alternate capabilities remain functional.
- Portable equipment that is expected to be unavailable for more than 90 days or expected to be unavailable during forecasted site specific external events (e.g., flooding) should be supplemented with alternate suitable equipment.
- The short duration of equipment unavailability, discussed above, does not constitute a loss of reasonable protection from a diverse storage location protection strategy perspective.
- 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.

The NRC staff finds that the licensee has adequately addressed equipment maintenance and testing activities associated with FLEX equipment because a maintenance and testing program has been established in accordance with NEI 12-06, Section 11.5. Further, the staff concludes that the licensee's unavailability controls conform to the provisions of NEI 12-06, Revision 2, with the clarification provided in Section 3.6.1.1 of this safety evaluation.

3.14 Alternatives to NEI 12-06, Revision 2

The licensee's FIP does not identify any alternatives to NEI 12-06, Revision 2. During the audit process the NRC staff identified that the three RPV makeup options listed in FIP Section 3.1.5 appear to connect and flow through a single train ("A" train RHR/RHRSW). This conflicts with the provisions of NEI 12-06, Revision 2, Table C-1, which states that connections should not be in the same train/division. This plan feature results in a common section of piping for all of the licensee's portable FLEX pump RPV makeup options. By letter dated September 28, 2017 [Reference 58], the licensee acknowledged not fully meeting the provisions of NEI 12-06, Table C-1, and provided an alternative justification.

In the alternative justification the licensee describes the routing of the impacted section of common piping. The piping section and associated valves are fully contained within protected structures and the system involved (RHR) is classified as a Class I system, with all the attendant qualifications of such a system. In addition, the licensee discusses the inherent ruggedness of the relevant piping and valves and evaluates the capability of the system both in terms of a design-basis earthquake, as well as an earthquake at the level of the reevaluated hazard under the NRC's 50.54(f) letter. The licensee also confirmed that the valves in the flowpath that would need to be opened are not double disk Anchor Darling gate valves that have been the subject of recent operating experience failures [Reference 56]. Further if a valve were not able to be opened, a flowpath to the opposite train of injection could be established via a cross-tie line. The NRC staff notes that procedures to accomplish this alignment, if needed, are in place.

The NRC staff reviewed the licensee's alternative justification. By having a section of common piping in the FLEX flowpath, the licensee loses some of the flexibility that conformance to NEI 12-06 was intended to provide for responding to potential beyond-design-basis events. However, the licensee's justification provides a basis to conclude that the subject piping would likely survive a beyond-design-basis seismic event, due to the margin associated with the inherently rugged components, especially considering the relatively low reevaluated hazard levels at Monticello. For other BDBEEs, the licensee has provided a sufficient basis to show that failure of the common piping is unlikely.

Thus, the NRC staff finds the licensee's proposed alternative to be acceptable.

3.15 Conclusions for Order EA-12-049

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

4.0 TECHNICAL EVALUATION OF ORDER EA-12-051

By letter dated February 28, 2013 [Reference 28], the licensee submitted an OIP for Monticello in response to Order EA-12-051. By email dated June 7, 2013 [Reference 29], the NRC staff sent a Request for Additional Information (RAI) to the licensee. The licensee provided a response to the RAI by letter dated July 12, 2013 [Reference 30]. By letter dated October 28, 2013 [Reference 31], the NRC staff issued an ISE and RAI to the licensee.

By letters dated August 28, 2013 [Reference 32], February 28, 2014 [Reference 33], August 28, 2014 [Reference 34], and February 24, 2015 [Reference 35], 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 SFP level instrumentation which will function following a BDBEE, including modifications necessary to support this implementation, pursuant to Order EA-12-051. By letter dated July 28, 2015 [Reference 37], the licensee reported that full compliance with the requirements of Order EA-12-051 was achieved.

The licensee has installed a SFP level instrumentation system designed by MOHR Test and Measurement, LLC (MOHR). The NRC staff reviewed the vendor's SFP level instrumentation system design specifications, calculations and analyses, test plans, and test reports. The staff issued an audit report regarding the MOHR system on August 27, 2014 [Reference 36].

The staff performed an onsite audit to review the implementation of SFP level instrumentation related to Order EA-12-051 at Monticello. The scope of the audit included verification of whether the: (a) site's seismic and environmental conditions are enveloped by the equipment qualifications, (b) equipment installation met the order requirements and vendor's recommendations, and (c) program features met the order requirements. By letter dated March 17, 2015 [Reference 22], the NRC issued an audit report on the licensee's progress.

4.1 Levels of Required Monitoring

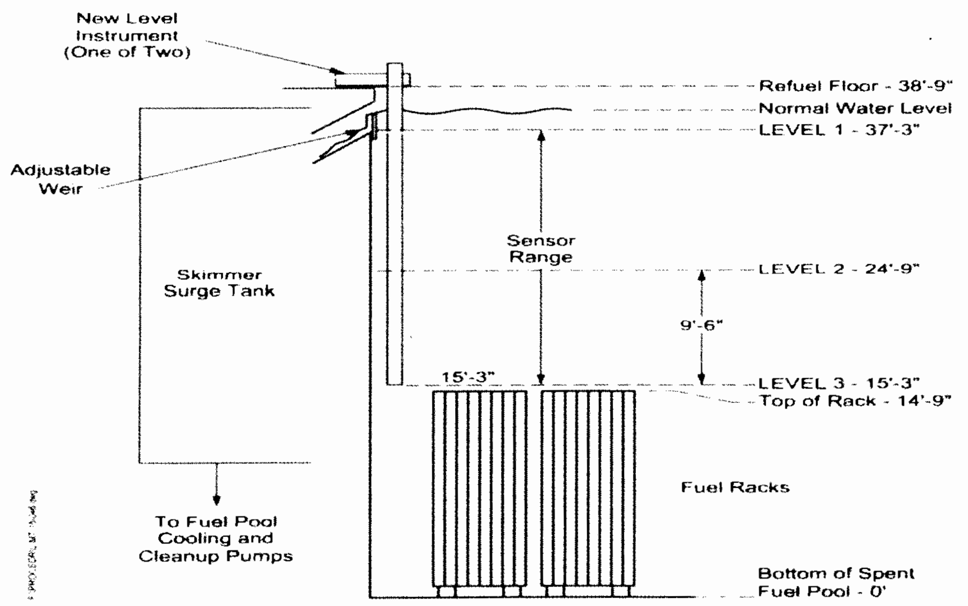
Attachment 2 of Order EA-12-051 states, in part:

> 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 the eighth six-month update for Order EA-12-049, dated February 20, 2017 [Reference 18], the licensee identified the SFP levels of monitoring as follows:

- Level 1 corresponds to 37 feet - 3 inches from the bottom of the SFP
- Level 2 corresponds to of 24 feet - 9 inches from the bottom of the SFP
- Level 3 corresponds to 15 feet - 3 inches from the bottom of the SFP

In the licensee's letter dated February 20, 2017, a sketch was provided depicting the SFP levels of monitoring. The sketch was titled "Figure 1-1 - Spent Fuel Pool Levels (Revised)," and is reproduced below:



Regarding to the Level 1 designation, in its OIP, the licensee stated that the level in the SFP is maintained at a constant level due to overflow weirs that spill into skimmer surge tanks. This configuration permits the total water volume to change within the skimmer surge tanks but it does not impact the SFP water level. The minimum level, without weirs, where the skimmer surge tank and the SFP become decoupled is 37 feet - 3 inches from the bottom of the SFP.

This level will be used for Level 1. In its RAI response letter dated July 12, 2013 [Reference 30], the licensee further stated that the water level in the skimmer surge tanks, not the water level in the SFP, provides the required net positive suction head (NPSH) to the fuel pool cooling and cleanup pumps. The fuel pool cooling and cleanup pumps are tripped on low level in the skimmer surge tank. The minimum SFP level that assures water is delivered to the fuel pool cooling and cleanup pumps is a SFP level that is above the height of the weir. Therefore, the SFP level that is adequate to support operation of the normal fuel pool cooling system is slightly above the height of the weir.

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

- Level 1: Level 1 is the level that is adequate for normal SFP cooling system operation and it is also adequate to ensure the required fuel pool cooling pump NPSH as the skimmer surge tanks supply the SFP cooling pumps. The licensee's chosen level meets the criteria described in NEI 12-02 for Level 1.

- Level 2: the Monticello Level 2 is consistent with the first of the two options described in NEI 12-02 for Level 2, which is 10 feet above the highest point of any fuel rack seated in the SFP.

- Level 3: Level 3 is above the highest point of any fuel rack seated in the SFP and thus corresponds to a level where fuel remains covered. Therefore, the licensee's Level 3 is consistent with NEI 12-02, which states that it corresponds nominally to the highest point of any fuel rack seated in the SFP.

Based on the evaluation above, the NRC staff finds that the licensee's selection of Levels 1, 2 and 3 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 Evaluation of Design Features

Order EA-12-051 requires that the SFP level instrumentation shall include specific design features, including specifications on the instruments, arrangement, mounting, qualification, independence, power supplies, accuracy, testing, and display. Below is the staff's assessment of the design features of the SFP level instrumentation.

4.2.1 Design Features: Instruments

The licensee's OIP states that the primary and backup instrument channel level sensing components would be permanently mounted in the SFP. The primary and backup instrument channels would be identical to and independent from each other, and would be based on a guided wave radar system. In its letter dated February 20, 2017 (described above), the licensee provided a sketch depicting the elevation view of the SFP with Level 1, 2, and 3 datum points and the level sensor's measurement range. The NRC staff notes that the measurement range specified for the licensee's instrumentation will cover Levels 1, 2, and 3.

The NRC staff finds that the licensee's design, with respect to the number of SFP instrument channels and instrument's measurement 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

The licensee's compliance letter states that the primary SFP level instrumentation sensor probe is located in the northeast corner of the SFP and the backup SFP level instrumentation sensor probe is located in the southeast corner. The primary channel display is in the control room and the backup channel display is located near the ASDS panel in the EFT Building. Cables are routed from the sensor probes to the read-out/display device location such that they ensure independence and redundancy of the channels.

Based on the licensee's description, and confirmed by a walkdown conducted during the onsite audit, the NRC staff concludes that there is sufficient channel separation between the primary and backup SFP level instrumentation channel's level probes, 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 finds that the licensee's arrangement for the SFP level instrumentation appears to be consistent with the NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.3 Design Features: Mounting

In its compliance letter dated July 28, 2015, the licensee stated that a calculation of the total loading on the bracket including design-basis seismic loads and hydrodynamic loads was performed. According to the licensee's letter, the structural members of the brackets are designed using a static equivalent force method. The vertical static equivalent force is then combined with other applicable forces and moments provided by the vendor. The licensee also stated that the bracket design meets Class I requirements in accordance with NEI 12-02 and includes use of an additional conservation factor of 50 percent greater than the site design basis seismic response spectra.

Related to the mounting design for the SFP level instrumentation electronics, the staff reviewed EC 23419, "Fukushima Response Spent Fuel Pool Instrumentation," Revision 0, during the audit process. The staff notes that Section 2.4 of the design description form states, in part, that the EFP-IL signal processor is a panel-mounted instrument with a NEMA 4X enclosure. The main (primary) instrument channel will be seismically mounted on the west control room wall. The backup instrument channel will be seismically mounted near the alternate shutdown panel in the EFT Building. Section 3.2.5.1 of EC 23419 further indicates that the SFP level instrumentation indicator and battery assembly is mounted to the wall via concrete expansion anchors. The envelop of the seismic horizontal accelerations for the applicable elevation of the control room and the EFT Building are 0.35 g and 0.32 g for rigid supports, respectively. The coaxial cable is routed in conduit in the Reactor Building.

During the onsite audit, the NRC staff reviewed the mounting specifications and seismic analyses for the SFP level instrumentation, including the methodology and design criteria used to estimate the total loading on the mounting devices. The staff also reviewed the design inputs and the methodology used to qualify the structural integrity of the affected structures for each of the SFP level instrumentation mounting attachments. Based on the review, the staff found the criteria established by the licensee adequately account for the appropriate structural loading conditions, including seismic and hydrodynamic loads.

Based on the licensee's compliance letter description, supplemented by the onsite audit review, the NRC staff finds that the licensee's mounting design appears to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.2.4 Design Features: Qualification

4.2.4.1 Augmented Quality Process

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

In its OIP, the licensee stated that the primary and backup instrument channels will be qualified through the use of an augmented quality assurance process that meets the provisions of JLD-ISG-2012-03 and NEI 12-02.

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

4.2.4.2 Equipment Reliability

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 components used 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.

The NRC staff reviewed the MOHR SFP level instrumentation's qualification and testing for temperature, humidity, radiation, shock and vibration, and seismic during the vendor audit. The staff also reviewed the anticipated Monticello environmental conditions during the onsite audit. Below is the staff's assessment of the equipment reliability of the Monticello SFP level instrumentation.

4.2.4.2.1 Radiation, Temperature, and Humidity

Spent Fuel Pool Area

According to the licensee's compliance letter, the SFP level instrumentation system equipment located in the SFP area has been qualified to withstand postulated accident conditions with an expected life of 40 years.

Outside of SFP Area

According to the licensee's compliance letter, the post BDBEE environment of the control boxes, electronics, and read-out and retransmitting devices was assessed with respect to temperature, humidity and radiation. This assessment includes the postulated BDB radiological and environmental conditions of the control room and the EFT Building where the primary and backup SFP level instrumentation electronics equipment and instrument displays are located, respectively. For the control room, temperature will not exceed 120°F during the first 48 hours of the event. The licensee also evaluated the EFT Building where the SFP backup level indication is installed (near the ASDS panel). For this area temporary ventilation when battery charging is being performed is provided as part of FLEX strategies in a BDBEE event.

During the audit process that staff reviewed the projected Monticello environmental conditions and the staff concludes that the equipment design limits envelop the anticipated conditions of radiation, temperature, and humidity during a postulated BDBEE (and post event) at the site. Further, based on the equipment environmental testing, the staff concludes that the licensee has demonstrated that the SFP instrumentation should maintain its functionality under the expected conditions.

4.2.4.2.2 Shock and Vibration

In its compliance letter, the licensee stated that the electronics of the SFP level instrumentation system equipment and associated batteries were tested to demonstrate general robustness. This testing included shock resistance for handling and transport, and vibration resistance appropriate for equipment in large power plants.

The NRC performed an audit at the MOHR facility that included a review of the shock and vibration testing of the SFP level instrumentation sensors and electronics components. During that audit, the staff found the testing protocol to be acceptable. Based on the licensee compliance letter statements and the vendor audit review, the staff concludes that, if implemented appropriately, the Monticello SFP level instrumentation should perform reliably with respect to shock and vibration.

4.2.4.2.3 Seismic

With regard to the SFP level instrumentation seismic qualification, the staff reviewed the licensee's design change document EC 23419, during the audit process. Section 4 of design description form states that the SFP level instrumentation equipment and components are non-safety related. However, it also states that the components are mounted seismically and that the MOHR instrumentation components are seismically qualified. Finally, the routing of cables and installation of equipment will be completed per applicable safety-related quality assurance requirements to ensure the equipment is mounted seismically.

The NRC staff review of the licensee's engineering change documentation concludes that the SFP level instrumentation was tested to seismic conditions that envelop Monticello's design basis maximum ground motion. In addition, Section 4.2.3 of this safety evaluation discussion provides further discussion of the SFP level instrumentation mounting details.

In summary, the NRC staff finds the Monticello SFP level instrumentation qualification process to be adequate. However, the staff has learned of operating experience at other nuclear facilities, in which the MOHR's SFP level instrumentation experienced failures of the filter coil (or choke). MOHR has determined the source of the failures is a miniature surface mount common-mode choke component used on the video and digicomp printed circuit boards within the EFP-IL Signal Processor. The vendor has developed and qualified substitute components that are less susceptible to transient electrical events. During the audit process the staff confirmed that the licensee had implemented the vendor recommended repair at Monticello for both SFP level instrumentation channels through the site corrective action and work order programs.

Based on the evaluation above, the NRC staff finds that the licensee's 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 compliance letter, the licensee described the SFP level instrument channel's physical independence as follows:

> The SFP level instrumentation system design includes two separate and independent channels; a primary instrument channel and a backup instrument channel... The normal power supply to the instruments is provided by different sources such that a loss of a distribution panel will not result in the loss of both channels... One indicator is located in the CR [control room] and the backup indicator is located near the ASDS panel in the EFT Building. Cables for each channel are routed in separate conduits and cable trays.

In its compliance letter, the licensee also indicates that the normal power supply to the instruments is provided by different busses such that a loss of a single bus will not result in the loss of both channels. During a BDBEE each channel has an independent battery system which will supply the channels with power for seven days in minimum power mode.

With the licensee's proposed design, the NRC staff notes that the loss of one level instrument channel would not affect the operation of other channel under BDBEE conditions. Further, the staff was able to conduct a walkdown of the instrument channel locations and routing during the audit process. Based on the licensee's compliance letter, confirmed by the audit walkdown, the NRC 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 compliance letter the licensee stated the following:

> Each level indicating channel is powered by an independent 120 VAC source. Each channel is provided with a battery back-up power supply capable of powering the channel for seven days... Each channel is powered from a different 480 V bus. Therefore, loss of any one 480 V bus does not result in loss of normal 120 VAC power for both instrument channels. On loss of normal 120 VAC power, each channel's UPS automatically transfers to a dedicated backup battery. If normal power is restored, the channel will automatically transfer back to the normal AC power. Each backup battery is maintained in a charged state

by a UPS. The batteries are sized to be capable of supporting for seven days of monitoring.

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

4.2.7 Design Features: Accuracy

In its compliance letter, the licensee stated that following:

> The absolute system accuracy is better than ±3 inches for all expected conditions. The instrument channel accuracy performance is approximately ±1% [percent] of span based on the sensitive range of the detector. This is conservative, bounding instrument channel accuracy with the vendor estimating expected instrument channel accuracy to be better than the above bounding accuracy. This accuracy is applicable for normal conditions and also the temperature, humidity, chemistry, and radiation levels expected for beyond BDB event conditions. Accuracy was validated by Factory Acceptance Testing.

Regarding restoration after loss of power the licensee's compliance letter states:

> The system automatically swaps to available power (backup battery power or external power source) when normal power is lost. Neither the source of power nor system restoration impact the accuracy.

The NEI 12-02 provisions for accuracy include stipulations that the instrument channels should maintain their accuracy following a power interruption or change in power source without recalibration. In addition, accuracy should consider the SFP conditions and trained personnel should be able to determine whether the actual SFP level exceeds the lower level of each indicating range (Levels 1, 2, and 3) without conflicting or ambiguous information. Based on the licensee's compliance letter, the NRC staff finds that the licensee has demonstrated that the SFP level instrumentation accuracy will be maintained during the expected normal and BDB conditions. Further, if the design is implemented properly, the instrument channels should maintain the designed accuracy following a change or interruption of power source without the need of recalibration.

Based on the licensee's compliance letter, the staff finds that the proposed instrument accuracy features 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.

4.2.8 Design Features: Testing

In its compliance letter dated July 28, 2015, the licensee indicated that MOHR's documentation provides a description of the capability and provisions the level sensing equipment has to enable periodic testing and calibration, including how this capability enables the equipment to be tested in-situ. The licensee also provided a description of channel cross-check, functional check, and calibration check features of the MOHR system, as installed at Monticello.

Based on the licensee's compliance letter, the NRC staff finds the licensee's proposed SFP instrumentation design allows for proper testing and calibration. Thus, the staff concludes that the testing features 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.

4.2.9 Design Features: Display

In its OIP, the licensee stated that the primary display will be located in the control room and will provide the operators an on-demand or continuous display of the SFP level. In its compliance letter, the licensee further stated that the backup SFP level instrumentation display is located near the ASDS panel on the third floor of the EFT Building.

Related to the accessibility of the display locations, in its compliance letter, the licensee stated that the backup SFP level instrumentation display, just like the ASDS panel, is easily accessible from the control room. Operators walk out of the control room to a corridor that leads to the EFT Building, up a stairway, and into the ASDS panel room. According to the licensee, this walk would take less than 10 minutes and the walking path is environmentally and radiologically mild.

Regarding the habitability of the backup display location, in its compliance letter, the licensee stated that the EFT Building is independent from the Reactor and Turbine Buildings. The EFT Building provides a safe enclosure and protection for the main components of the control room air conditioning system (including the emergency filtration train units for the air conditioning system) and for other safety-related equipment, in addition to housing the ASDS panel. According to the licensee, the EFT area has a mild environment and is independent from the secondary containment, the turbine building, and the radioactive waste building.

According to NEI 12-02, the display location shall have the following features: (1) an occupied area or promptly accessible area, (2) a location outside of the area surrounding the SFP floor, (3) a location inside a structure providing protection against adverse weather, and (4) a location outside of any very high radiation areas or locked high radiation areas. The staff notes that the licensee's description meets all of these provisions. Thus, the NRC staff finds that the licensee's proposed location and design of the SFP level instrumentation displays appear to be consistent with NEI 12-02 guidance, as endorsed by JLD-ISG-2012-03, and should adequately address the requirements of the order.

4.3 Evaluation of Programmatic Controls

4.3.1 Programmatic Controls: Training

In its OIP, the licensee stated that personnel will be trained in the use and the provision of alternate power to the primary and backup instrument channels, as determined by plant processes and procedures. Monticello's Systematic Approach to Training (SAT) will be used to identify the population to be trained, as well as 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. Thus, the NRC staff finds 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 for both the primary and backup instrument channels consistent with the requirements of NRC JLD-ISG-2012-03 and NEI 12-02. This will include procedures for the maintenance, operation, testing, calibration and normal/abnormal response of the primary and backup instrument channels. In its compliance letter, the licensee provided a listing and a description of the procedures applicable to the SFP level instrumentation design change. During the audit process the staff reviewed a sampling of these procedures. Based on the licensee's description and the audit review, the staff concludes that procedures have been developed for SFP level instrumentation testing, surveillance, calibration, test, operation, maintenance, and abnormal response, and that these procedures are consistent with the recommendations from the vendor.

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

4.3.3 Programmatic Controls: Testing and Calibration

In its compliance letter, the licensee described the testing and calibration program at Monticello. In summary, the licensee stated that periodic testing and calibration of the SFP level instrumentation was established in conjunction with the requirements of the vendor technical manuals. At a minimum, functional checks are performed at a frequency commensurate with vendor requirements. The SFP level instrumentation levels are verified daily at both instrument displays. Calibration checks are described in detail in the vendor operator's manual.

The applicable information is provided in plant procedures or preventive maintenance tasks. Formal calibration checks are recommended by the vendor on a two-year interval to demonstrate calibration to external National Institute of Standards and Technology traceable standards. Formal calibration check surveillance interval and timing are established in procedures consistent with vendor requirements. Finally, the SFP level instrumentation preventive maintenance performance is controlled through tasks in the Monticello PM program.

In its compliance letter, the licensee provided a description of the compensatory measures for the SFP level instrument channel(s) out-of-service. The licensee stated the following:

> The primary or backup 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 include the 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-functioning, then actions are initiated within 24 hours to restore one of the channels of instrumentation and to implement compensatory actions within 72 hours. Compensatory actions include the steps

necessary to ensure availability of normal alarms and increased direct visual monitoring of spent fuel pool level.

Based on the licensee's description, the staff concludes that the maintenance and test activities are consistent with MOHR recommendations. The staff also finds that the compensatory actions for non-functional SFP level instrumentation channels appear to be consistent with those recommended by NEI 12-02.

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

4.4 Conclusions for Order EA-12-051

In its compliance letter dated July 28, 2015 [Reference 37], the licensee stated that it met the requirements of Order EA-12-051 by following the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee's plans conform to the guidelines of NEI 12-02, as endorsed by JLD-ISG-2012-03. Based on the evaluations above, the NRC staff concludes that if the SFP level instrumentation is installed at Monticello according to the licensee's design, it should adequately address the requirements of Order EA-12-051.

5.0 CONCLUSION

In August 2013 the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit in November 2014 [Reference 22]. The licensee reached its final compliance date on May 11, 2017, for Order EA-12-049, and May 30, 2015, for Order EA-12-051, 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 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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SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT– 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. MF0923 AND MF0924; EPID NOS. L-2013-JLD-0015 AND L-2013-JLD-0016) DATED December 10, 2017

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