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

October 1, 2019

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

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NO. MF4376; EPID NO. L-2014-JLD-0052)

Dear Mr. Church:

On June 6, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13143A334), the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-13-109, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," to all Boiling Water Reactor (BWR) licensees with Mark I and Mark II primary containments. The order requirements are provided in Attachment 2 to the order and are divided into two parts to allow for a phased approach to implementation. The order required each licensee to submit an Overall Integrated Plan (OIP) for review that describes how compliance with the requirements for both phases of Order EA-13-109 would be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14183A412), Northern States Power Company (the licensee), doing business as Xcel Energy, submitted its Phase 1 OIP for Monticello Nuclear Generating Plant (Monticello) in response to Order EA-13-109. At 6-month intervals following the submittal of the Phase 1 OIP, the licensee submitted status reports on its progress in complying with Order EA-13-109 at Monticello, including the combined Phase 1 and Phase 2 OIP in its letter dated December 17, 2015 (ADAMS Accession No. ML15356A120). These status reports were required by the order and are listed in the enclosed safety evaluation. By letters dated May 27, 2014 (ADAMS Accession No. ML14126A545), and August 10, 2017 (ADAMS Accession No. ML17220A328), the NRC notified all BWR Mark I and Mark II licensees that the staff will be conducting audits of their implementation of Order EA-13-109 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated April 2, 2015 (Phase 1) (ADAMS Accession No. ML15082A167), September 6, 2016 (Phase 2) (ADAMS Accession No. ML16244A120), and April 10, 2018 (ADAMS Accession No. ML18094A804), the NRC issued Interim Staff Evaluations and an audit report, respectively, on the licensee's progress. By letter dated April 25, 2019 (ADAMS Accession No. ML19120A180), the licensee reported that Monticello is in full compliance with the requirements of Order EA-13-109, and submitted a Final Integrated Plan for Monticello.

The enclosed safety evaluation provides the results of the NRC staff's review of Monticello's hardened containment vent design and water management strategy for Monticello. The intent of the safety evaluation is to inform Monticello on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Order EA-13-109. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515-193, "Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions" (ADAMS Accession No. ML17249A105). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

If you have any questions, please contact Dr. Rajender Auluck, Senior Project Manager, Beyond-Design-Basis Engineering Branch, at 301-415-1025, or by e-mail at Rajender.Auluck@nrc.gov.

Sincerely,

for

Brett A. Titus, Acting Chief
Beyond-Design-Basis Engineering 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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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ORDER EA-13-109

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 at 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) during applicable severe accident conditions.

On June 6, 2013 [Reference 1], the NRC issued Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions." This order requires licensees to implement its requirements in two phases. In Phase 1, licensees of boiling-water reactors (BWRs) with Mark I and Mark II containments shall design and install a venting system that provides venting capability from the wetwell during severe accident conditions. In Phase 2, licensees of BWRs with Mark I and Mark II containments shall design and install a venting system that provides venting capability from the drywell under severe accident conditions, or, alternatively, those licensees shall develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions.

By letter dated June 30, 2014 [Reference 2], Northern States Power Company (the licensee), doing business as Xcel Energy, submitted its Phase 1 Overall Integrated Plan (OIP) for Monticello Nuclear Generating Plant (MNGP, Monticello) in response to Order EA-13-109. By letters dated December 16, 2014 [Reference 3], June 22, 2015 [Reference 4], December 17, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 17, 2016 [Reference 6], December 19, 2016 [Reference 7], June 14, 2017 [Reference 8], December 21, 2017 [Reference 9], June 26, 2018 [Reference 10], and December 17, 2018 [Reference 11], the licensee submitted 6-month updates to its OIP. By letters dated May 27, 2014 [Reference 12], and August 10, 2017 [Reference 13], the NRC notified all BWR Mark I and Mark II licensees that the staff will be conducting audits of their implementation of Order EA-13-109 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 14]. By letters dated April 2, 2015 (Phase 1) [Reference 15], September 6, 2016 (Phase 2) [Reference 16], and April 10, 2018 [Reference 17], the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated April 25, 2019 [Reference 18], the licensee reported that full compliance with the requirements of Order EA-13-109 was achieved and submitted its Final Integrated Plan (FIP).

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 improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a 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 19]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012 [Reference 20], 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", to the Commission. This paper included a proposal to order licensees to implement the installation of a reliable hardened containment venting system (HCVS) for Mark I and Mark II containments. As directed by the Commission in staff requirements memorandum (SRM)-SECY-12-0025 [Reference 21], the NRC staff issued Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents" [Reference 22], which required licensees to install a reliable HCVS for Mark I and Mark II containments.

While developing the requirements for Order EA-12-050, the NRC acknowledged that questions remained about maintaining containment integrity and limiting the release of radioactive materials if the venting systems were used during severe accident conditions. The NRC staff presented options to address these issues for Commission consideration in SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments" [Reference 23]. In the SRM for SECY-12-0157 [Reference 24], the Commission directed the staff to issue a modification to Order EA-12-050, requiring licensees with Mark I and Mark II containments to "upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions." The NRC staff held a series of public meetings following issuance of SRM SECY-12-0157 to engage stakeholders on revising the order. Accordingly, as directed by the Commission in SRM-SECY-12-0157, on June 6, 2013, the NRC staff issued Order EA-13-109.

Order EA-13-109 requires that BWRs with Mark I and Mark II containments have a reliable, severe-accident capable HCVS. Attachment 2 of the order provides specific requirements for implementation of the order. The order shall be implemented in two phases.

2.1 Order EA-13-109, Phase 1

For Phase 1, licensees of BWRs with Mark I and Mark II containments are required to design and install a venting system that provides venting capability from the wetwell during severe accident conditions. Severe accident conditions include the elevated temperatures, pressures, radiation levels, and combustible gas concentrations, such as hydrogen and carbon monoxide, associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris.

The NRC staff held several public meetings to provide additional clarifications on the order's requirements and comments on the proposed draft guidance prepared by the Nuclear Energy Institute (NEI) working group. On November 12, 2013 [Reference 25], NEI issued NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 0, to provide guidance to assist nuclear power reactor licensees with the identification of measures needed to comply with the requirements of Phase 1 of Order EA-13-109. The NRC staff reviewed NEI 13-02, Revision 0, and on November 14, 2013 [Reference 26], issued Japan Lessons-Learned Project Directorate (JLD) interim staff guidance (ISG) JLD-ISG-2013-02, "Compliance with Order EA-13-109, 'Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Performing under Severe Accident Conditions'", endorsing, in part, NEI 13-02, Revision 0, as an acceptable means of meeting the requirements of Phase 1 of Order EA-13-109, and on November 25, 2013, published a notice of its availability in the *Federal Register* (78 FR 70356).

2.2 Order EA-13-109, Phase 2

For Phase 2, licensees of BWRs with Mark I and Mark II containments are required to design and install a venting system that provides venting capability from the drywell under severe accident conditions, or, alternatively, to develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions.

The NRC staff, following a similar process, held several meetings with the public and stakeholders to review and provide comments on the proposed drafts prepared by the NEI working group to comply with the Phase 2 requirements of the order. On April 23, 2015 [Reference 27], NEI issued NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 1, to provide guidance to assist nuclear power reactor licensees with the identification of measures needed to comply with the requirements of Phase 2 of Order EA-13-109. The NRC staff reviewed NEI 13-02, Revision 1, and on April 29, 2015 [Reference 28], the NRC staff issued JLD-ISG-2015-01, "Compliance with Phase 2 of Order EA-13-109, 'Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Performing under Severe Accident Conditions'", endorsing, in part, NEI 13-02, Revision 1, as an acceptable means of meeting the requirements of Phase 2 of Order EA-13-109, and on April 7, 2015, published a notice of its availability in the *Federal Register* (80 FR 26303).

3.0 TECHNICAL EVALUATION OF ORDER EA-13-109, PHASE 1

Monticello is a single unit General Electric BWR site with a Mark I primary containment system. Containment integrity is maintained by controlling containment pressure using the HCVS. The HCVS is initiated using manual action from the primary operating station (POS) at the alternate shutdown system (ASDS) panel or remote operating station (ROS) at the appropriate time based on procedural guidance in response to plant conditions from observed or derived symptoms. The HCVS utilizes containment parameters of drywell pressure and wetwell water level from the ASDS panel instrumentation to monitor effectiveness of the venting actions. Vent operation is monitored by HCVS valve position, temperature, and effluent radiation levels. The HCVS motive force is monitored and has the capacity to operate for 24 hours with installed equipment. Replenishment of the motive force will be by use of portable equipment once the installed motive force is exhausted. Venting actions are capable of being maintained for a sustained period of at least 7 days.

3.1 HCVS Functional Requirements

3.1.1 Performance Objectives

Order EA-13-109 requires that the design and operation of the HCVS shall satisfy specific performance objectives including minimizing the reliance on operator actions and plant operators' exposure to occupational hazards such as extreme heat stress and radiological conditions, and accessibility and functionality of HCVS controls and indications under a broad range of plant conditions. Below is the staff's assessment of how the licensee's HCVS meets the performance objectives required by Order EA-13-109.

3.1.1.1 Operator Actions

Order EA-13-109, Attachment 2, Section 1.1.1 requires that the HCVS be designed to minimize the reliance on operator actions. Relevant guidance is found in NEI 13-02, Section 4.2.6 and HCVS-FAQ [Frequently Asked Questions]-01.

In its FIP, the licensee stated that the HCVS was designed to minimize the reliance on operator actions in response to the hazards identified in NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 2 [Reference 29], that are applicable to the plant site. Operator actions to initiate the HCVS vent path can be completed by plant personnel and include the capability for remote-manual initiation from the HCVS control station. A list of the remote manual actions performed by plant personnel to open the HCVS vent path are listed in Table 3-1, "HCVS Operator Actions," of the FIP. An HCVS extended loss of alternating current (ac) power (ELAP) Failure Evaluation Table (FIP Table 3-2), which shows alternate actions that can be performed, is also provided in the FIP.

The licensee also stated that permanently-installed electrical power and pneumatic supplies are available to support operation and monitoring of the HCVS for a minimum of 24 hours. No large portable equipment needs to be moved in the first 24 hours to operate the HCVS. After 24 hours, available personnel will be able to exchange depleted nitrogen bottles for full ones and transfer the HCVS control power to a direct current (dc) panel (0100) backed up by a FLEX generator. This provides electric controls and motive power for sustained operation of the HCVS for a minimum of 7 days. A supply of full nitrogen bottles is stored in each of the FLEX storage buildings (FSBs).

The NRC staff reviewed the HCVS Operator Actions Table, compared it with the information contained in NEI 13-02, and determined that these actions should minimize the reliance on operator actions. These actions are consistent with the type of actions described in NEI 13-02, Revision 1, as endorsed, in part, by JLD-ISG-2013-02 and JLD-ISG-2015-01, as an acceptable means for implementing applicable requirements of Order EA-13-109. The NRC staff also reviewed the HCVS Failure Evaluation Table and determined that the actions described adequately address all the failure modes listed in NEI 13-02, Revision 1, which include: loss of normal ac power; long-term loss of batteries; loss of normal pneumatic supply; loss of alternate pneumatic supply; and solenoid operated valve failure.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design should minimize the reliance on operator actions, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.1.2 Personnel Habitability – Environmental

Order EA-13-109, Attachment 2, Section 1.1.2 requires that the HCVS be designed to minimize plant operators' exposure to occupational hazards, such as extreme heat stress, while operating the HCVS. Relevant guidance is found in: NEI 13-02, Sections 4.2.5 and 6.1.1; NEI 13-02, Appendix I; and HCVS-FAQ-01.

In its FIP, the licensee stated that primary control of the HCVS is accomplished from the POS at the ASDS panel on the third floor of the emergency filtration train (EFT) building. Alternate control of the HCVS is accomplished from the remote operation station (ROS) at the ground floor elevation of the turbine building. The FIP states that the POS location will remain habitable under ELAP conditions based on an evaluation of the environmental conditions. Actions that may be taken to maintain the ASDS panel and ROS habitable include:

1. Providing fans or heaters as needed to maintain the EFT habitable for equipment and personnel. This equipment was included as a load in the 120V FLEX generator sizing calculations and is acceptable per Engineering Evaluation 28584.
2. Opening EFT doors to the outside (if required).

Table 2 of the FIP contains a thermal evaluation of all the operator actions that may be required to support HCVS operation. Calculation 16-055, "Monticello GOTHIC Analysis for the Hardened Containment Vent Project," Revision 0, demonstrates that the final design meets the order requirements to minimize the plant operators' exposure to occupational hazards.

The NRC staff audited Calculation 16-055. The calculation uses the Generation of Thermal-Hydraulic Information for Containment (GOTHIC) computer program to model various areas of the plant. The calculation uses input from Calculation 96-074, "Station Blackout (SBO) Heat-Up of the EFT Building," Revision 2. An initial temperature of 104 degrees Fahrenheit (°F) is assumed for the EFT building followed by an immediate jump to 120°F. Calculation 16-055 indicates that the temperature in the EFT building third floor (location of the primary operating station (POS)) would peak at 135°F in the summer at 12 hours. By 12 hours, supplemental ventilation will be installed per Procedure C.5-4503. The supplemental ventilation will maintain the temperature below 120°F. Figure 7.2-1 of the calculation indicates the EFT building third floor temperature varies between 110°F and 100°F with the daily diurnal temperature variation after supplemental ventilation is installed. The calculation assumes a daily peak outdoor temperature of 95°F. The NRC staff requested clarification that the high temperature in the POS would not hinder operator's ability to take the required actions. The licensee responded that the work in the POS is classified as light duty and consists of manipulating hand switches and periodic monitoring light indicators and indicator readings. Expected stay times are 10 minutes or less. Work in high temperature environments is controlled by the Monticello Safety Manual. In addition, the calculation assumes a very conservative step jump in temperature from a conservatively high initial room temperature of 104°F to 120°F. The large mass of concrete in the structure make such a step change unlikely. In winter, the procedure C.5-4503 instructs operators to use portable heaters as needed to maintain the temperature above 40°F. The NRC staff concurred that based on the calculation and procedural guidance environmental conditions should not prevent operators from performing their required actions.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to personnel habitability during severe accident conditions, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.1.3 Personnel Habitability – Radiological

Order EA-13-109, Attachment 2, Section 1.1.3 requires that the HCVS be designed to account for radiological conditions that would impede personnel actions needed for event response. Relevant guidance is found in: NEI 13-02, Sections 4.2.5 and 6.1.1; NEI 13-02, Appendices D, F, G and I; HCVS-FAQ-01, -07, -09 and -12; and HCVS-WP [White Paper]-02.

The licensee's Calculation 16-054, "MNGP HCVS Radiological Assessment," documents the dose assessment for designated areas inside the MNGP reactor building (outside of containment) and outside the MNGP reactor building caused by the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. Calculation 16-054 was performed using NRC-endorsed HCVS-WP-02 [Reference 30] and HCVS-FAQ-12 [Reference 31] methodologies. Consistent with the definition of sustained operations in NEI 13-02, Revision 1, the integrated whole-body gamma dose equivalent[1] due to HCVS operation over a 7-day period as determined in the licensee's dose calculation should not exceed 10 Roentgen equivalent man (rem)[2]. The calculated 7-day dose due to HCVS operation is a conservative maximum integrated radiation dose over a 7-day period with ELAP and fuel failure starting at reactor shutdown. For the sources considered and the methodology used in the calculation, the timing of HCVS vent operation or cycling of the vent will not create higher doses at personnel habitability and equipment locations (i.e., maximum doses determined in the calculation bound operational considerations for HCVS vent operation).

The licensee determined the expected dose rates in all locations requiring access following a beyond-design-basis ELAP. The licensee's evaluation indicates that for the areas requiring access in the early stages of the ELAP the expected dose rates would not be a limiting consideration. For those areas where expected dose rates would be elevated at later stages of the accident, the licensee has determined that the expected stay times would ensure that operations could be accomplished without exceeding the emergency response organization (ERO) emergency worker dose guidelines.

The licensee evaluated the maximum dose rates and 7-day integrated whole-body gamma dose equivalents for the main control room (MCR) which is the primary control location, and the ROS. In its FIP, the licensee states that the ROS location and the travel path to the

[1] For the purposes of calculating the personnel whole-body gamma dose equivalent (rem), it is assumed that the radiation units of Roentgen (R), radiation absorbed dose (rad), and Roentgen equivalent man (rem) are equivalent. The conversion from exposure in R to absorbed dose in rad is 0.874 in air and < 1 in soft tissue. For photons, 1 rad is equal to 1 rem. Therefore, it is conservative to report radiation exposure in units of R and to assume that 1 R = 1 rad = 1 rem.

[2] Although radiation may cause cancer at high doses and high dose rates, public health data do not absolutely establish the occurrence of cancer following exposure to low doses and dose rates below about 10,000 mrem (100 mSv). https://www.nrc.gov/about-nrc/radiation/health-effects/rad-exposure-cancer.html

ROS have been evaluated for habitability and accessibility during a severe accident. The licensee further states that during an accident, the distance and shielding combined with the short duration of actions required at the ROS show it to be an acceptable location for alternate control. The evaluation (as documented in Calculation 16-054) demonstrates that the integrated whole-body gamma dose equivalent to personnel occupying defined habitability locations (resulting from HCVS operation under beyond-design-basis severe accident conditions) should not exceed 10 rem.

The NRC staff notes that there are no explicit regulatory dose acceptance criteria for personnel performing emergency response actions during a beyond-design-basis severe accident. The Environmental Protection Agency (EPA) Protective Action Guides (PAG) Manual, EPA-400/R-16/001, "Protective Action Guides and Planning Guidance for Radiological Incidents," provides emergency worker dose guidelines. Table 3.1 of EPA-400/R-16/001 specifies a guideline of 10 rem for the protection of critical infrastructure necessary for public welfare, such as a power plant, and a value of 25 rem for lifesaving or for the protection of large populations. The NRC staff further notes that during an emergency response, areas requiring access will be actively monitored by health physics personnel to ensure that personnel doses are maintained as low as reasonably achievable.

The NRC staff audited the licensee's calculation of the expected radiological conditions to ensure that operating personnel can safely access and operate controls and support equipment. Based on the expected integrated whole-body dose equivalent in the MCR and ROS during the sustained operating period, the NRC staff agrees that the mission doses associated with actions taken to protect the public under beyond-design-basis severe accident conditions will not subject plant personnel to an undue risk from radiation exposure.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to personnel habitability during severe accident conditions, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.1.4 HCVS Controls and Indications

Order EA-13-109, Attachment 2, Section 1.1.4 requires that the HCVS controls and indications be accessible and functional under a range of plant conditions, including severe accident conditions, ELAP, and inadequate containment cooling. Relevant guidance is found in: NEI 13-02, Sections 4.1.3, 4.2.2, 4.2.3, 4.2.4, 4.2.5, and 6.1.1; NEI 13-02, Appendices F, G and I; and HCVS-FAQs-01 and -02.

Accessibility of the controls and indications for the environmental and radiological conditions are addressed in Sections 3.1.1.2 and 3.1.1.3 of this safety evaluation, respectively.

In Section III.B.1.1.4 of its FIP, the licensee stated that primary control of the HCVS is accomplished from the POS at the ASDS panel in the EFT building and that under the postulated scenarios of Order EA-13-109, the POS is adequately protected from excessive radiation dose. The licensee also stated that alternate control of the HCVS is accomplished from the ROS located in the turbine elevation on elevation 931'. The licensee stated the ROS location is in an area evaluated to be accessible before and during a severe accident. The licensee also provided, in Table 1 of its FIP, a list of the controls and indications that are or may be required to operate the HCVS during a severe accident, including the locations, anticipated environmental conditions, and the environmental conditions (temperature and radiation) to which each component is qualified.

The NRC staff reviewed the FIP, including the response in Section III.B.1.1.4 of the FIP and examined the information provided in Table 1. The NRC staff determined that the controls and indications appear to be consistent with the NEI 13-02 guidance. The NRC staff also confirmed the environmental qualification information in Table 1 of the FIP, as well as the seismic qualification of the controls and indications equipment through audit reviews of MNGP Calculations 16-054, "MNGP HCVS Radiological Assessment," Revision 0A and 16-055, "GOTHIC Analysis for the Hardened Containment Vent Project," Revision 0. The NRC staff noted that the Regulatory Guide (RG) 1.97 instruments for drywell pressure (not including the indicators) and wetwell level did not include some qualification information in Table 1, but are considered acceptable, in accordance with the NEI 13-02 guidance, based on the original qualification for severe accident conditions.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to accessibility and functionality of the HCVS controls and indications during severe accident conditions, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2 Design Features

Order EA-13-109 requires that the HCVS shall include specific design features, including specifications of the vent characteristics, vent path and discharge, control panel, power and pneumatic supply sources, inadvertent actuation prevention, HCVS monitoring, equipment operability, and hydrogen control. Below is the staff's assessment of how the licensee's HCVS meets the performance objectives required by Order EA-13-109.

3.1.2.1 Vent Characteristics

Order EA-13-109, Attachment 2, Section 1.2.1 requires that the HCVS has the capacity to vent the steam/energy equivalent of one percent of licensed/rated thermal power (unless a lower value is justified by analyses), and be able to restore and then maintain containment pressure below the primary containment design pressure and the primary containment pressure limit. Relevant guidance is found in NEI 13-02, Section 4.1.1.

The licensee performed Calculation 16-019, "Monticello Hardened Containment Vent System (HCVS) Capacity Analysis and Verification of Suppression Pool Decay Heat Capacity," Revision 1, which provides the verification of one percent power flow capacity at the design pressure (56 pounds per square inch (psig) in the drywell), which is lower than the primary containment pressure limit (PCPL) (62 psig). The calculation also verifies the decay heat absorbing capacity of the suppression pool. This calculation models all the piping elbows, valves and other components using a nodalized series of piping components. Since the piping consists of 8-inch and 10-inch diameter pipe sections, both are modeled. The model is input to RELAP5 computer code which is used to simulate transient two-phase flow conditions in piping systems. The code also looks for flow choking effects. The minimum flow at design pressure to pass the energy equivalent of one percent reactor thermal power is 75,718 pounds mass per hour (lbm/hr). Calculation 16-019 verifies that the piping can pass greater than one percent flow. The calculation demonstrates that containment is maintained below the design pressure once the vent is opened, even if it is not opened until PCPL.

The NRC staff reviewed the information provided and audited Calculation 16-019. The calculation determined that one percent of the licensed thermal power (2004 megawatts thermal (MWt)) venting requirement is 75,718 lbm/hr at 62 psig. The steady state venting capacity at a torus pressure of 47.9 psig (maximum design pressure in the drywell and the differential pressure between the drywell and wetwell with the torus completely full of water, is 79,737 lbm/hr (5.3 percent flow margin to one percent thermal power requirement). Flow varies from roughly 20,000 lbm/hr at 5 psig to 90,000 lbm/hr at 55 psig. Based on the evaluation, the HCVS vent design appears to have the capacity to vent one percent of rated thermal power during ELAP and severe accident conditions with margin.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design characteristics, if implemented appropriately, appear to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.2 Vent Path and Discharge

Order EA-13-109, Attachment 2, Section 1.2.2 requires that the HCVS discharge the effluent to a release point above main plant structures. Relevant guidance is found in: NEI 13-02, Section 4.1.5; NEI 13-02, Appendix H; and HCVS-FAQ-04.

The NRC staff evaluated the HCVS vent path and the location of the discharge. The vent taps off the top of the torus gas space with an 8-inch pipe. The pipe contains two primary containment isolation valves and a rupture disk. The vent exits the reactor building through the high-pressure coolant injection pump room roof. The vent traverses up the exterior of the building and discharges approximately five feet above the reactor building roof parapet. The NRC staff's review indicates that this appears to be consistent with the guidance provided in HCVS-FAQ-04.

The release point discharges away from emergency ventilation system intake and exhaust openings, main control room location, location of HCVS portable equipment, access routes required following an ELAP and BDBEE, and emergency response.

Guidance document NEI 13-02, Section 5.1.1.6, provides guidance that missile impacts are to be considered for portions of the HCVS. The NRC-endorsed NEI white paper, HCVS-WP-04, "Tornado Missile Evaluation for HCVS Components 30 Feet Above Grade," Revision 0 [Reference 32], provides a risk-informed approach to evaluate the threat posed to exposed portions of the HCVS by wind-borne missiles. The white paper concludes that the HCVS is unlikely to be damaged in a manner that prevents containment venting by wind-generated missiles coincident with an ELAP or loss of normal access to the ultimate heat sink (UHS) for plants that are enveloped by the assumptions in the white paper.

The licensee evaluated the vent pipe robustness with respect to wind-borne missiles against the assumptions contained in HCVS-WP-04. This evaluation demonstrated that the pipe was robust with respect to external missiles per HCVS-WP-04 in that:

1. The portion of exposed piping has been protected to an elevation of 30' above grade by the erection of a tornado missile barrier.

2. The exposed piping greater than 30' above grade has the following characteristics:

a. The total vent pipe exposed area is approximately 143 square-feet which is less than 300 square feet used in the guidance.
b. The exposed portion of the pipe is made of welded steel, designed to ASME B31.1, and contains no small tubing susceptible to missiles.
c. There are no obvious sources of missiles located in the proximity of the exposed HCVS components.

3. The HCVS at MNGP is designed to withstand the external events applicable to the site. Therefore, there is no need to provide for repairing or bypassing a damaged section of the vent piping.

4. Monticello was screened out for hurricanes, due to its location near the center of the North American continent.

The NRC staff audited evaluation EE-26081-01, "Reasonable Protection Evaluation Grade for HCVS Tornado Missile Barrier," along with EC 26083, "Hardened Containment Venting System NRC Order EA-13-109 Phase 1." The NRC staff agrees that the HCVS pipe is adequately protected against seismic or tornado events.

Based on the evaluation above, the NRC staff concludes that the licensee's location and design of the HCVS vent path and discharge, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.3 Unintended Cross Flow of Vented Fluids

Order EA-13-109, Attachment 2, Section 1.2.3 requires that the HCVS include design features to minimize unintended cross flow of vented fluids within a unit and between units on the site. Relevant guidance is found in: NEI 13-02, Sections 4.1.2, 4.1.4, and 4.1.6; and HCVS-FAQ-05.

In its FIP, the licensee described Monticello as a single-unit site and there are no other units on the site for the cross flow of vented fluids. The design of the HCVS is such that it is a completely independent system that runs from the torus to the roof of the reactor building.

Except for the rupture disk pneumatic supply, there are no interconnecting systems to the HCVS. A check valve at the rupture disk pneumatic supply connection prevents backflow to the ROS. The pneumatic supply line is normally isolated with manual and solenoid valves when not in use. In addition to the pneumatic supply connection check valve the NRC staff also note that primary containment isolation valves (PCIVs) are provided near the torus connection. These valves are subject to leak testing under the guidance of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," test program as part of the containment boundary. The NRC staff audited the information provided and agrees that the use of primary containment isolation valves appears to be acceptable for prevention of inadvertent cross-flow of vented fluids and consistent with the guidance provided in HCVS-FAQ-05.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design limits the potential for unintended cross flow of vented fluids and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.4 Control Panels

Order EA-13-109, Attachment 2, Section 1.2.4 requires that the HCVS be designed to be manually operated during sustained operations from a control panel located in the MCR or a remote but readily accessible location. Relevant guidance is found in NEI 13-02, Sections 4.2.2, 4.2.4, 4.2.5, 5.1, and 6.1; NEI 13-02, Appendices A and H; and HCVS-FAQs-01 and -08.

In its FIP, the licensee stated that the existing wetwell vent will allow initiating and then operating and monitoring from the ASDS panel on the third floor of the EFT building. The NRC staff notes that this location was identified as the POS in Section III.B.1.2.4 of the Monticello FIP. Table 1 of the FIP contains a list of the HCVS instrumentation and controls components including their location and qualification information. The NRC staff reviewed Section III.B.1.2.4 and confirmed these statements by comparing the instrumentation and controls component locations provided in Table 1 of the FIP.

Based on the evaluation above, the NRC staff concludes that the licensee's location and design of the HCVS control panels, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.5 Manual Operation

Order EA-13-109, Attachment 2, Section 1.2.5 requires that the HCVS, in addition to meeting the requirements of Section 1.2.4, be capable of manual operation (e.g., reach-rod with hand wheel or manual operation of pneumatic supply valves from a shielded location), which is accessible to plant operators during sustained operations. Relevant guidance is found in NEI 13-02, Section 4.2.3 and in HCVS-FAQs-01, -03, -08, and -09.

In its FIP, the licensee stated that to meet the requirement for an alternate means of operation, a readily accessible alternate location, called the ROS was added. The location for the ROS is on a concrete wall just south of the B alternate nitrogen bottle station, accessed through DOOR-26 (931' Turbine Building East). The ROS is shielded from the HCVS vent pipe by intervening structures, including the path to the MCR. The ROS contains manually operated valves that supply pneumatics to the HCVS flow path valve actuators and rupture disk so that the HCVS may be opened without power to the valve actuator solenoids and regardless of any containment isolation signals that may be actuated. This provides a diverse method of valve operation and improves system reliability.

The controls available at the ROS location are accessible and functional under a range of plant conditions including: severe accident conditions with due consideration to source term and dose impact on operator exposure; ELAP; inadequate containment cooling; and loss of reactor building ventilation. Table 1 of the FIP contains an evaluation of all the required controls and instruments that are required for severe accident response and demonstrates that all these controls and instruments will be functional during a loss of ac power and severe accident. Table 2 of the FIP contains a summary of thermal and radiological evaluations of all the operator actions that may be required to support HCVS operation during a loss of ac power and severe accident. The licensee's evaluations conclude that these actions will be possible without undue hazard to the operators. These evaluations demonstrate that the design meets the requirement to be manually operated from a remote, but readily accessible location during sustained operation. Attachment 6 of the FIP contains a site layout showing the location of these HCVS actions. The NRC staff audited the pertinent plant drawings and evaluation documents. The NRC staff's audit confirmed that the actions appear to be consistent with the guidance.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design allows for manual operation, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.6 Power and Pneumatic Supply Sources

Order EA-13-109, Attachment 2, Section 1.2.6 requires that the HCVS be capable of operating with dedicated and permanently installed equipment for at least 24 hours following the loss of normal power or loss of normal pneumatic supplies to air operated components during an ELAP. Relevant guidance is found in: NEI 13-02, Sections 2.5, 4.2.2, 4.2.4, 4.2.6, and 6.1; NEI 13-02, Appendix A; HCVS-FAQ-02; and HCVS-WPs-01 and -02.

Pneumatic Sources Analysis

For the first 24 hours following the ELAP event, the motive force for the actuation of the two PCIVs and the rupturing of the disc will be supplied by two nitrogen gas bottles. The nitrogen gas bottles are stationed on elevation 931' of the East turbine building. The ROS is south of the B alternate nitrogen bottle station and is shielded from the HCVS vent pipe by intervening structures, including the path to the MCR. The controls available at the ROS location are accessible and functional under a range of plant conditions, including severe accident conditions with due consideration to source term and dose impact on operator exposure, ELAP, inadequate containment cooling, and loss of ventilation.

The licensee determined the required pneumatic supply storage volume and supply pressure set point required to operate the PCIV actuation and disc rupture for 24 hours following a loss of normal pneumatic supplies during an ELAP in Calculation 16-011, "Calculation of HPV System Dedicated Nitrogen Supply and Pressure Requirements," Revision 0A. The required pressure for total HCVS operation is calculated at around 95 psig for PCIV actuation and 150 psig for rupturing the disc. The licensee's calculation determined that two nitrogen bottles, filled at the maximum capacity of 2640 psig, will provide sufficient capacity for eight cycles of the PCIVs and disc rupture for 24 hours following an ELAP. This pressure includes an allowance for leakage. The NRC staff audited the calculation and confirmed that there should be sufficient pneumatic supply available to provide motive force to operate the HCVS system for 24 hours following a loss of normal pneumatic supplies during an ELAP.

Power Source Analysis

In its FIP, the licensee stated that during the first 24 hours of an ELAP event, Monticello would rely on a new dedicated battery and battery charger with sufficient capacity to supply HCVS loads. The 125-volt (V) dc (Vdc) HCVS battery, two (normal and backup) battery chargers, distribution panel, and manual transfer switch are located in the emergency filtration train (EFT) building on the third floor. The HCVS battery and battery charger are installed where they are protected from applicable hazards. Exide Technologies manufactured the HCVS battery.

The HCVS battery is model GNB Absolyte GP 50G07 with a nominal capacity of 104 ampere hours (Ah). The HCVS battery has a minimum capacity capable of providing power for 24 hours without recharging. During the audit period, the licensee provided the NRC staff with an evaluation for the HCVS battery/battery charger sizing requirements including incorporation into the FLEX diesel generator (DG) loading calculation.

The NRC staff audited licensee Calculation 16-006, "Hard Pipe Vent D8 Battery HCVS 125VDC Battery Calculation," Revision 1, which verified the capability of the HCVS battery to supply power to the required loads during the first phase of the Monticello venting strategy for an ELAP. The HCVS battery was sized in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 485-1983, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Generating Stations and Substations," which is endorsed by RG 1.212, "Sizing of Large Lead-Acid Storage Batteries," published in 2008. The licensee's calculation identified the required loads and their associated ratings (current and minimum system operating voltage). The licensee's battery sizing calculation also showed that the minimum calculated terminal voltage at each device is above the minimum voltage required during the 24-hour discharge cycle. Therefore, the Monticello HCVS battery should have sufficient capacity to supply power for at least 24 hours.

The licensee's strategy includes transitioning power to the Division II, 250 Vdc battery (D6) within 24 hours, which would be energized by its battery chargers powered by a FLEX DG. The licensee's strategy relies on one of two portable 200-kilowatt (kW) 480 Volt alternating current (Vac) FLEX DGs. The 480 Vac FLEX DG would provide power to the HCVS load in addition of loads addressed under Order EA-12-049.

The NRC staff also audited licensee engineering evaluation EE 23964, "FLEX 480 V Diesel Generator Sizing" under Order EA-12-049. The evaluation shows that the loading for the 200 kW FLEX DG is 163.8 kW which includes the Division II, 250 Vdc battery charger. Therefore, sufficient margin exist on 200 kW FLEX DG to power HCVS loads.

<u>Electrical Connection Points</u>

The licensee's strategy to supply power to HCVS components requires using a combination of permanently installed and portable components. Staging and connecting the 200 kW FLEX DG were addressed under Order EA-12-049. Licensee procedure C.5-4453, "Energize Hard Pipe Vent During SBO," Revision 0, provides guidance to transfer electrical power for HCVS loads from D8 (125 Vdc HCVS Battery) to the Division II, 250 Vdc battery.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design allows for reliable operation with dedicated and permanently installed equipment, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.7 Prevention of Inadvertent Actuation

Order EA-13-109, Attachment 2, Section 1.2.7 requires that the HCVS include means to prevent inadvertent actuation. Relevant guidance is found in NEI 13-02, Section 4.2.1.

In its FIP, the licensee stated that emergency operating procedures (EOPs) provide clear guidance that the HCVS is not to be used to defeat containment integrity during any design basis transients and accidents. In addition, the HCVS was designed to provide features to prevent inadvertent actuation due to equipment malfunction or operator error. Also, these protections are designed such that any credited containment accident pressure (CAP) that would provide net positive suction head to the emergency core cooling system (ECCS) pumps will be available (inclusive of a design basis loss-of-coolant accident. The licensee credits CAP to maintain sufficient net positive suction head (NPSH) for ECCS pumps, core spray, and residual heat removal. Therefore, it is essential to prevent inadvertent actuation of the HCVS to ensure that the CAP can be maintained. During ELAP the ECCS pumps will not have normal power available.

At Monticello, inadvertent actuation prevention of the HCVS is accomplished by the following physical design features. Two containment isolation valves in series powered from different divisions and key lock switches prevent inadvertent operator action. The backup pneumatic supply is controlled by maintaining normally closed valves at the ROS. The NRC staff's audit confirmed that the licensee's design is consistent with the guidance and appears to preclude inadvertent actuation.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to prevention of inadvertent actuation, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.8 Monitoring of HCVS

Order EA-13-109, Attachment 2, Section 1.2.8 requires that the HCVS include means to monitor the status of the vent system (e.g. valve position indication) from the control panel required by Section 1.2.4. In addition, Order EA-13-109 requires that the monitoring system be designed for sustained operation during an ELAP. Relevant guidance is found in NEI 13-02, Section 4.2.2; and HCVS-FAQs-01, -08, and -09.

The NRC staff reviewed the following channels documented in Table 1 of the FIP that support HCVS operation: HCVS effluent temperature; HCVS battery voltage; HCVS effluent radiation; HCVS valve position; N_2 pressure (mechanical); drywell pressure; and wetwell level. The NRC staff notes that drywell pressure and wetwell level are declared Monticello post-accident monitoring (PAM) variables as described in RG 1.97 and the existing qualification of these channels is considered acceptable for compliance with Order EA-13-109 in accordance with the guidance in NEI 13-02, Appendix C, Section C.8.1. The NRC staff also reviewed FIP Section III.B.1.2.8 and determined that the HCVS instrumentation appears to be adequate to support HCVS venting operations and is capable of performing its intended function during ELAP and severe accident conditions.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design allows for the monitoring of key HCVS instrumentation, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.9 Monitoring of Effluent Discharge

Order EA-13-109, Attachment 2, Section 1.2.9 requires that the HCVS include means to monitor the effluent discharge for radioactivity that may be released from operation of the HCVS. In addition, Order EA-13-109 requires that the monitoring system provide indication from the control panel required by Section 1.2.4 and be designed for sustained operation during an ELAP. Relevant guidance is found in: NEI 13-02, Section 4.2.4; and HCVS-FAQs-08 and -09.

The NRC staff reviewed the following channels documented in Table 1 of the FIP which support monitoring of HCVS effluent: HCVS valve position; HCVS effluent temperature; and HCVS effluent radiation. The NRC staff found that effluent radiation monitor provides sufficient range to adequately indicate effluent discharge radiation levels.

In Section III.B.1.2.9 and Table 1 of its FIP, the licensee stated that the radiation monitor uses an ion chamber detector and that the radiation monitor detector is fully qualified for the expected environment at the vent pipe during accident conditions, and the process and control module is qualified for the environment at the POS in the EFT. The NRC staff audited the qualification summary information provided in Table 1 of the FIP and found that it appeared to meet the guidance.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design allows for the monitoring of effluent discharge, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.10 Equipment Operability (Environmental/Radiological)

Order EA-13-109, Attachment 2, Section 1.2.10 requires that the HCVS be designed to withstand and remain functional during severe accident conditions, including containment pressure, temperature, and radiation while venting steam, hydrogen, and other non-condensable gases and aerosols. The design is not required to exceed the current capability of the limiting containment components. Relevant guidance is found in: NEI 13-02, Sections 2.3, 2.4, 4.1.1, 5.1 and 5.2; NEI 13-02 Appendix I; and HCVS-WP-02.

Environmental

The FLEX diesel-driven severe accident water addition (SAWA) pump and FLEX DG will be staged outside so they will not be adversely impacted by a loss of ventilation.

The ASDS panel at the POS, HCVS battery, battery charger, and supporting equipment are permanently installed in the EFT building on the third floor. As discussed above in Section 3.1.1.2, the licensee performed Calculation 16-055, which predicts the temperature profile on the EFT building third floor following an ELAP. The licensee determined that performing compensatory ventilation actions (opening doors, and establishing portable ventilation or heaters) on the EFT building third floor will maintain temperatures in the area below 120°F. Licensee procedure C.5-4503, "EFT Ventilation During FLEX Conditions," Revision 0, directs operators to open doors and establish portable ventilation in the EFT building.

The Division II, 250 Vdc battery is installed in the EFT building on the first floor. The NRC staff audited licensee Calculation 15-017, "PAB/EFT Battery Rooms Heat-up During a SBO and ELAP," Revision 0, under Order EA-12-049 compliance and documented in the NRC staff's safety evaluation [Reference 37] that the licensee has developed a plan that, if implemented appropriately, should maintain or restore equipment functionality in the EFT building on the first floor following a BDBEE. The licensee determined that the maximum temperature in the battery room was 90.2°F. Licensee procedure C.5-4503, 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 concurs with the licensee's calculations that show the EFT building first floor will remain below the maximum temperature limit (122°F) of the Division II, 250 Vdc batteries, as specified by the battery manufacturer (C&D Technologies) and the third floor will remain below the maximum temperature limit (120°F) of the HCVS batteries, as specified by the battery manufacturer (Exide Technologies). Furthermore, based on temperature remaining below 120°F (the temperature limit for electronic equipment to be able to survive indefinitely, identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, as endorsed by NRC RG 1.155), the NRC staff believes that other electrical equipment located at POS should not be adversely impacted by the loss of ventilation as a result of an ELAP event with the HCVS in operation. Therefore, the NRC staff concurs that the HCVS equipment should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Radiological

The licensee's Calculation 16-054, "MNGP HCVS Radiological Assessment," documents the dose assessment for both personnel habitability and equipment locations associated with event response to a postulated ELAP condition. The NRC staff audited Calculation 16-054 and notes that the licensee used conservative assumptions to bound the peak dose rates for the analyzed areas. For the sources considered and the methodology used in the dose calculation, the timing of HCVS vent operation or cycling of the vent will not create higher doses at personnel habitability and equipment locations (i.e., maximum doses determined in the calculation bound operational considerations for HCVS vent operation). The NRC staff's audit confirmed that the anticipated severe accident radiological conditions will not preclude the operation of necessary equipment or result in an undue risk to personnel from radiation exposure.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to equipment operability during severe accident conditions, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.11 Hydrogen Combustible Control

Order EA-13-109, Attachment 2, Section 1.2.11 requires that the HCVS be designed and operated to ensure the flammability limits of gases passing through the system are not reached; otherwise, the system shall be designed to withstand dynamic loading resulting from hydrogen deflagration and detonation. Relevant guidance is found in: NEI 13-02, Sections 4.1.7, 4.1.7.1, and 4.1.7.2; NEI 13-02, Appendix H; and HCVS-WP-03.

Guidance document NEI 13-02, Section 4.1.7 provides guidance for the protection from flammable gas deflagration/detonation in the HCVS. The NEI issued white paper HCVS-WP-03, "Hydrogen/Carbon Monoxide Control Measures," Revision 1, endorsed by the NRC [Reference 34], which provides methods to address control of flammable gases. One of the acceptable methods described in the white paper is the installation of a check valve at or near the end of the vent stack to restrict the ingress of air to the vent pipe when venting stops and steam condenses (Option 5).

In its FIP, the licensee stated that to prevent a detonable mixture from developing in the pipe, a check valve is installed near the top of the pipe in accordance with HCVS-WP-03.

This valve will open on venting but will close to prevent air from migrating back into the pipe after a period of venting. The check valve is installed and tested to ensure that it limits back-leakage to preclude a detonable mixture from occurring in the case venting is stopped prior to the establishment of alternate reliable containment heat removal. The NRC staff audit confirmed the design appears to be consistent with Option 5 of the white paper HCVS-WP-03 and that the use of a check valve in conjunction with the HCVS venting strategy should meet the requirements to prevent a detonable mixture from developing in the pipe.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design ensures that the flammability limits of gases passing through the system are not reached, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.12 Hydrogen Migration and Ingress

Order EA-13-109, Attachment 2, Section 1.2.12 requires that the HCVS be designed to minimize the potential for hydrogen gas migration and ingress into the reactor building or other buildings. Relevant guidance is found in NEI 13-02, Section 4.1.6; NEI 13-02, Appendix H; HCVS-FAQ-05; and HCVS-WP-03.

As discussed in Section 3.2.1.2, the rupture disk pneumatic supply is the only interfacing mechanical system on the HCVS flow path which could lead to potential hydrogen gas migration and ingress into the reactor building or other buildings. The pneumatic supply is separated from the HCVS by solenoid valves and a check valve. In addition, the pneumatic supply connects to the HCVS through a 3/8-inch diameter hose. Any leakage through the pneumatic supply will be very small relative to the volume of the building. The NRC staff's audit confirmed that the design appears to be consistent with the guidance and meets the design requirements to minimize the potential of hydrogen gas migration into other buildings.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design minimizes the potential for hydrogen gas migration and ingress, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.13 HCVS Operation/Testing/Inspection/Maintenance

Order EA-13-109, Attachment 2, Section 1.2.13 requires that the HCVS include features and provisions for the operation, testing, inspection and maintenance adequate to ensure that reliable function and capability are maintained. Relevant guidance is found in NEI 13-02, Sections 5.4 and 6.2; and HCVS-FAQs-05 and -06.

In the Monticello FIP, Table 3-3 includes testing and inspection requirements for HCVS components. The NRC staff reviewed Table 3-3 and confirmed that it is consistent with Section 6.2.4 of NEI 13-02, Revision 1. Implementation of these testing and inspection requirements for the HCVS will ensure reliable operation of the systems.

In its FIP, the licensee stated that the maintenance program was developed using the guidance provided in NEI 13-02, Sections 5.4 and 6.2, and it utilizes the standard Electric Power Research Institute (EPRI) industry preventive maintenance process for the maintenance calibration and testing for the HCVS components. The NRC staff reviewed the information provided and confirmed that the licensee has implemented adequate programs for operation, testing, inspection and maintenance of the HCVS.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design allows for operation, testing, inspection, and maintenance, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.2 HCVS QUALITY STANDARDS

3.2.1 Component Qualifications

Order EA-13-109, Attachment 2, Section 2.1 requires that the HCVS vent path up to and including the second containment isolation barrier be designed consistent with the design basis of the plant. Items in this path include piping, piping supports, containment isolation valves, containment isolation valve actuators and containment isolation valve position indication components. Relevant guidance is found in NEI 13-02, Section 5.3.

In its FIP, the licensee stated that the HCVS piping upstream of and including the outboard containment isolation valve (AO-4540) and penetrations were not modified for order compliance so that they continue to be designed consistent with the design basis of primary containment including pressure, temperature, radiation, and seismic loads. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators, and containment isolation valve position indication components. The hardened vent piping between the wetwell and the reactor building roof is designed to 62 psig and 309°F.

Guidance document NEI 13-02 suggests a 350°F value for HCVS design temperature based on the highest PCPL among the Mark I and II plants. The Monticello PCPL is 62 psig with a corresponding saturation temperature of 309°F. Monticello used a design value of 309°F for the vent piping, corresponding to the saturation temperature for the MNGP PCPL of 62 psig. Thus, the temperature of 309°F will be retained as the pipe design temperature. This lower HCVS design temperature is adequate for component qualifications, since it is acceptable (per the guidance in NEI 13-02 Rev. 1, Section 2.4.3.1) to assume saturation conditions in containment.

The HCVS components downstream of the outboard containment isolation valve, including piping and supports, electrical power supply, valve actuator pneumatic supply, and instrumentation (local and remote) components, have been designed and analyzed to conform to the requirements consistent with the applicable design codes for the plant and to ensure functionality following a design basis earthquake. This includes environmental qualification consistent with expected conditions at the equipment location.

Table 1 of the FIP contains a list of components, controls and instruments required to operate the HCVS, their qualification limits, and a summary of the expected environmental conditions. All instruments are fully qualified for the expected seismic conditions so that they will remain functional following a seismic event. The NRC staff reviewed Table 1 and confirmed that the components required for HCVS venting are designed to remain functional following a design basis earthquake.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to component qualifications, if implemented appropriately, appears to be consistent with

NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.2.2 Component Reliability and Rugged Performance

Order EA-13-109, Attachment 2, Section 2.2 requires that all other HCVS components be designed for reliable and rugged performance, capable of ensuring HCVS functionality following a seismic event. These items include electrical power supply, valve actuator pneumatic supply, and instrumentation (local and remote) components. Relevant guidance is found in NEI 13-02, Sections 5.2 and 5.3.

In its FIP, the licensee stated that HCVS components downstream of the outboard containment isolation valve and components that interface with the HCVS are routed in seismically-qualified structures or supported from seismically-qualified structure(s) and that all instruments are fully qualified for the expected seismic conditions so that they will remain functional following a seismic event.

As part of the NRC staff's audit, the NRC staff requested information verifying that the new containment isolation valves, relied upon for the HCVS will open under the maximum expected differential pressure during beyond-design-basis and severe accident wetwell venting. The licensee performed Calculation 03-088, "AOV Component Calculation, Hard Pipe Vent Valves, AO-4539 and AO-4540," Revision 5, which discusses the valve/actuator information for PCIVs. The calculation demonstrates that valves will open under a maximum differential pressure of 76.7 pounds per square inch differential (psid). The NRC staff's audit noted that the full opening maximum torque is 252 foot-pounds and the corresponding actuator capacity at that required valve torque is 304 foot-pounds. Therefore, the PCIVs should open under the maximum expected differential pressure during beyond-design-basis and severe accident wetwell venting.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design, with respect to component reliability and rugged performance, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.3 Conclusions for Order EA-13-109, Phase 1

Based on its review, the NRC staff concludes that the licensee has developed guidance and a HCVS design that, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.0 TECHNICAL EVALUATION OF ORDER EA-13-109, PHASE 2

As stated above in Section 2.2, Order EA-13-109 provides two options to comply with the Phase 2 order requirements. Monticello has elected the option to develop and implement a reliable containment venting strategy that makes it unlikely the licensee would need to vent from the containment drywell before alternate reliable containment heat removal and pressure control is reestablished.

For this method of compliance, the order requires licensees to meet the following:

- The strategy making it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions shall be part of the overall accident management plan for Mark I and Mark II containments;
- The licensee shall provide supporting documentation demonstrating that containment failure as a result of overpressure can be prevented without a drywell vent during severe accident conditions; and,
- Implementation of the strategy shall include licensees preparing the necessary procedures, defining and fulfilling functional requirements for installed or portable equipment (e.g. pumps and valves), and installing the needed instrumentation.

Relevant guidance is found in NEI 13-02, Sections 4, 5 and 6; and Appendices C, D, and I.

4.1 Severe Accident Water Addition (SAWA)

The licensee plans to use the FLEX (SAWA) pump to provide SAWA flow into the reactor pressure vessel (RPV). Flow control for SAWA will be performed at the SAWA pump along with instrumentation and procedures to ensure that the wetwell vent is not submerged. Once SAWA flow is initiated, operators will have to monitor and maintain SAWA flow and ensure refueling of the diesel-driven equipment as necessary. In its FIP, the licensee states that the operator locations for deployment and operation of the SAWA equipment that are external to the reactor building are either shielded from direct exposure to the vent pipe or are a significant distance from the vent pipe so that dose will be maintained below ERO exposure guidelines.

4.1.1 Staff Evaluation

4.1.1.1 Flow Path

The SAWA injection flow path starts with the FLEX pump taking suction from the discharge canal and goes through the flexible discharge hose to the connection point at valve RHRSW-68 in the turbine building through a flow meter. The RHRSW-68 valve is connected to the residual heat removal (RHR) system, then to the reactor pressure vessel (RPV) through the low-pressure coolant injection (LPCI) line. This SAWA injection path is qualified for the external hazards in addition to severe accident conditions. The hoses and pumps used for SAWA flow are stored in the FLEX storage building and Warehouse 6, which are both protected from all external hazards.

4.1.1.2 SAWA Pump

In its FIP, the licensee states that the strategy is to use two redundant portable diesel-driven pumps, in which one pump will be used for FLEX and SAWA strategies. The two FLEX pumps are trailer-mounted and are capable of withstanding seismic events. The two FLEX pumps are each capable of 300 gallons per minute (gpm) flow. The two FLEX pumps are stored in the FLEX storage building and Warehouse 6 respectively, where they are protected from all applicable external hazards. The initial SAWA flow will be injected into the RPV within 8 hours of the loss of injection. In its FIP, the licensee described the hydraulic analysis performed to demonstrate the capability of the two portable FLEX pumps to provide the required SAWA flow.

The NRC staff audited Calculation 15-004, "Monticello Flex Pump Simultaneous SFP/RPV Flow," Revision 0, which determined that the required SAWA flow rate of 285 gpm was within the capacity of at least one FLEX pump.

The NRC staff audited the flow rates and pressures evaluated in the hydraulic analyses and confirmed that the equipment can provide the needed flow. Based on the NRC staff's audit of the FLEX pumping capabilities, as described in the above hydraulic analysis and the FIP, it appears that the licensee has demonstrated that the portable FLEX pumps should perform as intended to support SAWA flow.

4.1.1.3 SAWA Analysis of Flow Rates and Timing

The licensee developed the overall accident management plan for Monticello from the BWR Owner's Group (BWROG) emergency procedure guidelines and severe accident guidelines (EPG/SAG) and NEI 13-02, Appendix I. The SAWA/SAWM [Severe Accident Water Management] implementing procedures are integrated into the MNGP severe accident management guidelines (SAMGs). The EPG/SAG Revision 3, when implemented with emergency procedures committee generic issue 1314, allows throttling of SAWA valves to protect containment while maintaining the wetwell vent in service. The SAMG flow charts direct use of the hardened vent as well as SAWA/SAWM when the appropriate plant conditions have been reached.

The licensee used NEI 12-06, Appendix E to validate the FLEX water system pumps used for SAWA can be deployed and commence injection in less than 8 hours. The studies referenced in NEI 13-02 demonstrated that establishing flow within 8 hours will protect containment. Guidance document NEI 13-02, Appendix I, establishes an initial water addition rate of 500 gpm based on EPRI Technical Report 3002003301, "Technical Basis for Severe-Accident Mitigating Strategies." The initial SAWA flow rate at Monticello will be at least 285 gpm based on the site's rated thermal power compared to the reference power level in NEI 13-02. After roughly 4 hours, during which the maximum flow rate is maintained, the SAWA flow will be reduced. The reduction in flow rate and the timing of the reduction will be based on stabilization of the containment parameters of drywell pressure and torus level.

The licensee used the referenced plant analysis included in NEI 13-02, Revision 1, information from EPRI Technical Report 3002003301, and MNGP-specific parameters to demonstrate that SAWA flow could be reduced to 57 gpm after 4 hours of initial SAWA flow rate and containment would remain protected. At some point, if wetwell level begins to rise, indicating that the SAWA flow is greater than the steaming rate due to containment heat load, SAWA flow can be further reduced as directed by the SAMGs.

In its FIP, the licensee stated that the torus vent was designed and installed to meet NEI 13-02, Revision 1, guidance and is sized to prevent containment overpressure under severe accident conditions. The licensee will follow the guidance (flow rate and timing) for SAWA described in BWROG-TP-15-008, "Severe Accident Water Addition Timing," [Reference 35] and BWROG-TP-15-011 "Severe Accident Water Management" [Reference 36]. The wetwell vent will be opened prior to exceeding the PCPL value of 62 psig. The licensee also referenced analysis included in BWROG-TP-15-008, which demonstrates adding water to the reactor vessel within 8 hours of the onset of the event will limit the peak containment drywell temperature, significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the containment from the suppression chamber.

The NRC staff audited the information referenced above. Guidance document NEI 13-02, uses an initial SAWA flow of 500 gpm reduced to 100 gpm after four hours. The NRC staff noted that the licensee determined plant-specific flow rates using the ratio of Monticello licensed thermal power (2004 MWt) to that of the reference plant (3,514 MW!) in the EPRI Technical Report 3002003301, "Technical Basis for Severe-Accident Mitigating Strategies." This is consistent with NEI 13-02, Section 4.1.1.2.

4.1.2 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has developed SAWA guidance that should ensure protection of the containment during severe accident conditions following an ELAP event, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.2 Severe Accident Water Management

The licensee's strategy to preclude the necessity for installing a hardened drywell vent at Monticello is to implement the containment venting strategy utilizing SAWA and SAWM. This strategy consists of the use of the Phase 1 torus vent and SAWA hardware to implement a water management strategy that will preserve the torus vent path until alternate reliable containment heat removal can be established. The SAWA system consists of a FLEX pump injecting into the RPV. The overall strategy consists of flow control by throttling valves at the FLEX (SAWA) pump along with instrumentation and procedures to ensure that the wetwell vent is not submerged (SAWM). Water from the SAWA (FLEX) pump will be routed through the RHR system. The RHR connection allows the water to flow in the RPV. Throttling valves and flow meters will be used to control water flow to maintain wetwell availability. Procedures have been issued to implement this strategy including Revision 3 to the SAMG. This strategy has been shown via Modular Accident Analysis Program (MAAP) analysis to protect containment without requiring a drywell vent for at least seven days which is the guidance from NEI 13-02 for the period of sustained operation.

4.2.1 Staff Evaluation

4.2.1.1 Available Freeboard Use

In its FIP, the licensee states that the freeboard between normal wetwell level of 910' and 925' elevation provides approximately 728,812 gallons of water volume before the wetwell level instrument would be off scale high. A diagram of the available freeboard is shown on Attachment 1 to the FIP.

Generic assessment BWROG-TP15-011 provides the principles of SAWM to preserve the wetwell vent for a minimum of 7 days. After containment parameters are stabilized with SAWA flow, SAWA flow will be reduced to a point where containment pressure will remain low while wetwell level is stable or very slowly rising. For Monticello, the SAWA/SAWM design flow rates (285 gpm at 8 hours followed by 57 gpm from 12 hours to 168 hours) and available freeboard volume (described above) are bounded by the values utilized in the BWROG-TP-15-011 reference plant analysis that demonstrates the success of the SAWA/SAWM strategy.

As shown in evaluation 608000000102, "SAWA Flowrates and Torus Water Levels," the wetwell level will not reach the wetwell vent for at least seven days. The NRC staff audited the information provided and agrees that starting the water addition at the higher flow rate and throttling back after approximately 4-hours will not increase the suppression pool level to a point that could block the suppression chamber HCVS opening before operators can take additional actions to maintain containment integrity.

4.2.1.2 Strategy Time Line

As noted in Section 4.1.1.3, "SAWA Analysis of Flow Rates and Timing," the SAWA flow is based on evaluation 608000000102 and BWROG-TP-15-011 to demonstrate that throttling SAWA flow after containment parameters have stabilized, in conjunction with venting containment through the torus vent will result in a stable or slowly rising torus level. The references demonstrate that, for the scenario analyzed, wetwell level will remain below the upper range of the wetwell level instrument, and below the wetwell vent pipe for greater than the seven days of sustained operation allowing significant time for restoration of alternate containment pressure control and heat removal. The NRC staff concurs that the SAWM approach should provide operators sufficient time to reduce the water flow rate and to maintain wetwell venting capability. The strategy is based on BWROG generic assessments in BWROG-TP-15-008 and BWROG-TP-15-011.

As noted above, BWROG-TP-15-008 demonstrates adding water to the reactor vessel within 8-hours of the onset of the event will limit the peak containment drywell temperature significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the suppression chamber through the suppression pool. Technical Paper BWROG-TP-011 demonstrates that starting water addition at a high rate of flow and throttling after approximately 4-hours will not increase the suppression pool level to that which could block the suppression chamber HCVS.

4.2.2 <u>Conclusions</u>

Based on this evaluation, the NRC staff concludes that the licensee has developed SAWM guidance that should make it unlikely that the licensee would need to vent from the containment drywell during severe accident conditions following an ELAP event, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.3 <u>SAWA/SAWM Motive Force</u>

4.3.1 <u>Staff Evaluation</u>

4.3.1.1 SAWA Pump Power Source

As described in Section 4.1, the licensee plans to use one of two portable diesel-driven FLEX pumps to provide SAWA flow. Operators will refuel the FLEX pump and DGs in accordance with Order EA-12-049 procedures using fuel oil from the installed emergency diesel generator (EDG) fuel oil storage tanks. Procedure A.8-06.03, "Refueling Emergency Portable Diesel Powered Equipment," Revision 3, directs operators to refuel the portable FLEX pumps from the onsite EDG fuel oil storage tanks. In its FIP, the licensee states that refueling will be accomplished in areas that are qualified for severe accident scenarios.

4.3.1.2 DG Loading Calculation for SAWA/SAWM Equipment

In its FIP, the licensee list drywell pressure, wetwell level, and the portable SAWA flow meter, as instruments required for SAWA and SAWM implementation. The wetwell level and drywell pressure are used for HCVS venting operation. These instruments are powered by the Class 1E station batteries until the FLEX DG is deployed and available. The SAWA flow meter is an electromagnetic flow meter that has an internal battery that provides 3 to 6 years of operation.

The NRC staff audited licensee dc coping calculation 14-102, "250V D6 Battery FLEX Coping Time Analysis," Revision 0, which verified the capability of the Class 1E station batteries to supply power to the required loads (e.g. wetwell level and drywell pressure) during the first phase of the Monticello FLEX mitigation strategy plan for an ELAP event. The NRC staff also audited licensee Engineering Evaluation 23964, which verified that the 200 kW is adequate to support the other SAWA/SAWM electrical loads which include the LPCI Swing Bus. The NRC staff confirmed that the Class 1E station batteries and 200 kW FLEX DGs should have sufficient capacity and capability to supply the necessary SAWA/SAWM loads during an ELAP event.

4.3.2 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has established the necessary motive force capable to implement the water management strategy during severe accident conditions following an ELAP event, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.4 <u>SAWA/SAWM Instrumentation</u>

4.4.1 <u>Staff Evaluation</u>

4.4.1.1 SAWA/SAWM Instruments

In Section IV.C.10.2 of its FIP, the licensee stated that the instrumentation needed to implement the SAWA/SAWM strategy are wetwell level, drywell pressure and SAWA flow meter. The NRC staff found that wetwell level and drywell pressure are existing RG 1.97 instruments that were designed and qualified for severe accident conditions. The licensee stated, in Table 1 of its FIP, that SAWA flow instrument range is 38 to 1914 gpm or 50 to 2000 gpm, depending on which one is deployed. The SAWA flow instrument range appears to be consistent with the licensee's strategy. The NRC staff reviewed Section IV.C.10.1, Section IV.C.10.2, and Table 1 of the FIP and found the instruments appear to be consistent with the NEI 13-02 guidance.

4.4.1.2 SAWA Instruments and Guidance

In Section IV.C.10.2 of its FIP, the licensee stated that the drywell pressure and wetwell level instruments used to monitor the condition of containment are pressure and differential pressure detectors that are safety-related and qualified for post-accident use. The licensee strategy may also make use of drywell temperature. The licensee also stated that SAMG strategies will evaluate and use drywell temperature indication if available consistent with the symptom based approach.

In Section IV.C.10.2 of its FIP, the licensee stated that the SAWA flow meter is an in-line flow meter powered by its own batteries and installed as part of setting up the SAWA connection.

The NRC staff reviewed the FIP, including Table I and Section IV.C.10.2 and found that the licensees response appears to be consistent with the guidance. The NRC staff notes that NEI 13-02 Revision 1 Section C.8.3 clarifies that drywell temperature is not required, but may provide further information for the operations staff to evaluate plant conditions under severe accident and provide confirmation to adjust SAWA flow rates.

4.4.1.3 Qualification of SAWA/SAWM Instruments

In Section IV.C.10.3 of its FIP, the licensee stated drywell pressure and wetwell level are declared Monticello PAM variables as described in RG 1.97 and the existing qualification of these channels is considered acceptable for compliance with Order EA-13-109 in accordance with the guidance in NEI 13-02, Appendix C, Section C.8.1. The NRC staff verified the RG 1.97 variables in the Monticello Final Safety Analysis Report. The NRC staff notes that the licensee clarified in the FIP that the drywell pressure indicator in the MCR is not RG 1.97 qualified, but is qualified for the mild environment in the MCR.

In its FIP, the licensee stated that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. Furthermore, since the pump is deployed inside the turbine building 931' east area near the connection point, it is protected from weather and high radiation near the vent pipe.

The licensee stated in Table 1 of its FIP that anticipated temperature at this location is greater than 0°F and the qualification temperature is -4°F - +140°F or -40°F – +176°F depending on which one is deployed. The licensee also stated, in Table 1 of its FIP, that the flow meter, located inside the turbine building 931' East, has an anticipated local radiation level of 0.186 R/hr peak/15.5 R/hr total for 7 days and is qualified for 50 sieverts or 5 E+3 R. The NRC staff determined the SAWA flow meter appears to be qualified for the anticipated environment.

4.4.2 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has in place, the appropriate instrumentation capable to implement the water management strategy during severe accident conditions following an ELAP event, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.5 <u>SAWA/SAWM Severe Accident Considerations</u>

4.5.1 <u>Staff Evaluation</u>

4.5.1.1 Severe Accident Effect on SAWA Pump and Flowpath

In its FIP, the licensee stated that the FLEX pumps are stored in the FSB and Warehouse 6 and will be operated from outside the reactor building, on the opposite side of the reactor building from the vent pipe. Therefore, there will be no significant issues with radiation dose rates at the SAWA pump control location and there will be no significant dose to the SAWA pump.

In its FIP, the licensee stated that the SAWA flow path inside the reactor building consists of steel piping that will be unaffected by the anticipated radiation dose rates.

The NRC staff audited the information and agrees that the SAWA flow path will not be adversely affected by radiation effects due to the severe accident conditions.

4.5.1.2 Severe Accident Effect on SAWA/SAWM Instruments

The MNGP SAWA strategy relies on three instruments: wetwell level; containment pressure; and SAWA flow. Containment pressure and wetwell level are declared PAM variables as described in RG 1.97 and the existing qualification of these channels is considered acceptable for compliance with Order EA-13-109 in accordance with the guidance in NEI 13-02, Appendix C, Section C.8.1.

In its FIP, the licensee states that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. Based on this information, the NRC staff agrees that the SAWA/SAWM instruments should not be adversely affected by radiation effects due to severe accident conditions.

4.5.1.3 Severe Accident Effect on Personnel Actions

According to the FIP when implementing SAWA, personnel must deploy a portable diesel generator and connect it to the LPCI swing bus. Personnel must also deploy and connect a portable diesel pump (PDP) to the FLEX connection at RHRSW-68 in the turbine building. Personnel would then use the reenergized swing bus to align the LPCI injection motor-operated valve (MOV) in the reactor building and operate the PDP to inject water. Table 2 of the FIP describes the actions within the first 7 hours. These actions including access routes outside the reactor building that will be performed after the first use of the vent during severe accident conditions (assumed to be 7 hours per HCVS-FAQ-12) are located such that they are either shielded from direct exposure to the vent line or are a significant distance from the vent line, so that expected dose is maintained below the ERO exposure guidelines.

The licensee performed GOTHIC calculations of the temperature response of the locations of instruments, control elements, and personnel actions during the ELAP event. These calculations determined that the equipment will function and the areas to be occupied by personnel will remain habitable for the period of sustained operations.

Table 2 of the FIP provides a list of SAWA/SAWM operator actions as well as an evaluation of each for suitability during a severe accident. Attachment 6 to the FIP shows the approximate locations of the actions.

After the SAWA flow path is aligned inside the reactor building, the operators can control SAWA/SAWM in the turbine building and observe the necessary instruments from the POS and the turbine building. Environmental conditions in the POS was evaluated earlier in Section 3.1.1.2, "Personnel Habitability – Environmental." The SAWA pump and monitoring equipment can all be operated from the POS, the turbine building or from outside at ground level. The MNGP FLEX response ensures that the SAWA pump, FLEX generators and other equipment can all be run for a sustained period by refueling. The NRC staff reviewed the projected environmental conditions for monitoring and operating the SAWA/SAWM strategy and concludes that the environmental conditions will not prevent operators from performing required actions to implement that plan.

The licensee performed Calculation 16-054, "MNGP HCVS Radiological Assessment," which documents the dose assessment for designated areas inside the MNGP reactor building (outside of containment) and outside the MNGP reactor building caused by FLEX activities and the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. This assessment used conservative assumptions to determine the expected dose rates in all areas that may require access during a beyond-design-basis ELAP. As stated in Section 3.1.1.3, Personnel Habitability - Radiological, the NRC staff agrees, based on the audit of the licensee's detailed evaluation, that mission doses associated with actions taken to protect the public under beyond-design-basis severe accident conditions will not subject plant personnel to an undue risk from radiation exposure.

4.5.2 Conclusions

Based on this evaluation, the NRC staff concludes that the licensee has considered the severe accident effects on the water management strategy and that the operation of components and instrumentation should not be adversely affected, and the performance of personnel actions should not be impeded, during severe accident conditions following an ELAP event. The NRC staff further concludes that the water management strategy, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

4.6 Conclusions for Order EA-13-109, Phase 2

Based on its review, the NRC staff concludes that the licensee has developed guidance and a water management strategy that, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

5.0 HCVS/SAWA/SAWM PROGRAMMATIC CONTROLS

5.1 Procedures

Order EA-13-109, Attachment 2, Section 3.1 requires that the licensee develop, implement, and maintain procedures necessary for the safe operation of the HCVS. Furthermore, Order EA-13-109 requires that procedures be established for system operations when normal and backup power is available, and during an ELAP. Relevant guidance is found in NEI 13-02, Sections 6.1.2 and 6.1.2.1.

In its FIP, the licensee states that a site-specific program and procedures were developed following the guidance provided in NEI 13-02, Sections 6.1.2, 6.1.3, and 6.2. They address the use and storage of portable equipment including routes for transportation from the storage locations to deployment areas. In addition, the procedures have been established for system operations when normal and backup power is available, and during ELAP conditions. The FIP also states that provisions have been established for out-of-service requirements of the HCVS and the compensatory measures. In the FIP, Section V.B provides specific time frames for out-of-service requirements for HCVS functionality.

The FIP also provides a list of key areas where either new procedures were developed, or existing procedures were revised. The NRC staff audited the overall procedures and programs developed, including the list of key components included, and noted that they appear to be consistent with the guidance found in NEI 13-02, Revision 1. The NRC staff determined that procedures developed appear to be in accordance with existing industry protocols. The provisions for out-of-service requirements appear to reflect consideration of the probability of an ELAP requiring severe accident venting and the consequences of a failure to vent under such conditions.

Based on the evaluation above, the NRC staff concludes that the licensee's procedures for HCVS/SAWA/SAWM operation, if implemented appropriately, appear to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the Order.

5.2 Training

Order EA-13-109, Attachment 2, Section 3.2 requires that the licensee train appropriate personnel in the use of the HCVS. Furthermore, Order EA-13-109 requires that the training include system operations when normal and backup power is available, and during an ELAP. Relevant guidance is found in NEI 13-02, Section 6.1.3.

In its FIP, the licensee stated that all personnel expected to perform direct execution of the HCVS/SAWA/SAWM actions will receive necessary training. The training plan has been developed per the guidance provided in NEI 13-02, Section 6.1.3, and will be refreshed on a periodic basis as changes occur to the HCVS actions, systems, or strategies. In addition, training content and frequency follows the systems approach to training process. The NRC staff reviewed the information provided in the FIP and confirmed that the training plan is consistent with the established systems approach to training process.

Based on the evaluation above, the NRC staff concludes that the licensee's plan to train personnel in the operation, maintenance, testing, and inspection of the HCVS design and water management strategy, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

6.0 CONCLUSION

In June 2014, the NRC staff started audits of the licensee's progress in complying with Order EA-13-109. The staff issued an ISE for implementation of Phase 1 requirements on April 2, 2015 [Reference 15], an ISE for implementation of Phase 2 requirements on September 6, 2016 [Reference 16], and an audit report on the licensee's responses to the ISE open items on April 10, 2018 [Reference 17]. The licensee reached its final compliance date on April 25, 2019, and in letter dated April 25, 2019 [Reference 17], has declared that Monticello is in compliance with the order and submitted its FIP.

Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance that includes the safe operation of the HCVS design and a water management strategy that, if implemented appropriately, should adequately address the requirements of Order EA-13-109.

7.0 REFERENCES

1. Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," June 6, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13143A321)

2. Letter from Monticello to NRC, "Monticello Nuclear Generating Plant – Phase 1 Overall Integrated Plan in Response to June 6, 2013, Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions Phase 1 (Order Number EA-13-109)," dated June 30, 2014 (ADAMS Accession No. ML14183A412)

3. Letter from Monticello to NRC, "First Six-Month Status Report For Phase 1 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated December 16, 2014 (ADAMS Accession No. ML14353A215)

4. Letter from Monticello to NRC, "Second Six-Month Status Report For Phase 1 Overall Integrated Plan in Response to June 6, 2013, Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated June 22, 2015 (ADAMS Accession No. ML15173A176)

5. Letter from Monticello to NRC, "Monticello Nuclear Generating Plant – Phase 1 (Updated) and Phase 2 Overall Integrated Plan in Response to June 6, 2013, Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order EA-13-109)," dated December 17, 2015 (ADAMS Accession No. ML15356A120)

6. Letter from Monticello to NRC, "Fourth Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013, Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated June 17, 2016 (ADAMS Accession No. ML16169A309)

7. Letter from Monticello to NRC, "Fifth Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated December 19, 2016 (ADAMS Accession No. ML16354A666)

8. Letter from Monticello to NRC, "Sixth Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated June 14, 2017 (ADAMS Accession No. ML17166A051)

9. Letter from Monticello to NRC, "Seventh Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying

Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated December 21, 2017 (ADAMS Accession No. ML17355A508)

10. Letter from Monticello to NRC, "Eighth Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated June 26, 2018 (ADAMS Accession No. ML18177A422)

11. Letter from Monticello to NRC, "Ninth Six-Month Status Report For Phases 1 and 2 Overall Integrated Plan in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated December 17, 2018 (ADAMS Accession No. ML18352A254)

12. Nuclear Regulatory Commission Audits of Licensee Responses to Phase 1 of Order EA-13-109 to Modify Licenses With Regard To Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions, dated May 27, 2014 (ADAMS Accession No. ML14126A545)

13. Nuclear Regulatory Commission Audits of Licensee Responses to Phase 2 of Order EA-13-109 to Modify Licenses With Regard To Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions, dated August 10, 2017 (ADAMS Accession No. ML17220A328)

14. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195)

15. Letter from NRC to Monticello, "Monticello Nuclear Generating Plant – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Phase 1 of Order EA-13-109 (Severe Accident Capable Hardened Vents)," dated April 2, 2015 (ADAMS Accession No. ML15082A167)

16. Letter from NRC to Monticello, "Monticello Nuclear Generating Plant – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Phase 2 of Order EA-13-109 (Severe Accident Capable Hardened Vents)," dated September 6, 2016 (ADAMS Accession No. ML16244A120)

17. Letter from NRC to Monticello, "Monticello Nuclear Generating Plant – Report for The Audit of Licensee Responses to Interim Staff Evaluations Open Items Related to NRC Order EA-13-109 to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated April 10, 2018 (ADAMS Accession No. ML18094A804)

18. Letter from Monticello to NRC, "Monticello Nuclear Generating Plant, Report of Full Compliance with Phase 1 and Phase 2 of June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)," dated April 25, 2019 (ADAMS Accession No. ML19120A180)

19. SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 (ADAMS Accession No. ML111861807)

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

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

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Date: October 1, 2019

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NO. MF4376; EPID NO. L-2014-JLD-0052) DATED: October 1, 2019

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