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

April 10, 2018

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

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - 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 (CAC NO. MF4376; EPID 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 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 licensees to submit for review overall integrated plans (OIPs) that describe how compliance with the requirements for both phases of Order EA-13-109 will be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14183A412), Northern States Power Company - Minnesota (NSPM, the licensee) submitted its Phase 1 OIP for Monticello Nuclear Generating Plant (MNGP, Monticello). By letters dated December 16, 2014, June 22, 2015, December 17, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 17, 2016, December 19, 2016, June 14, 2017, and December 21, 2017 (ADAMS Accession Nos. ML14353A215, ML15173A176, ML15356A120, ML16169A309, ML16354A666, ML17166A051, and ML17355A508, respectively), the licensee submitted its 6-month updates to the OIP. The NRC staff reviewed the information provided by the licensee and issued interim staff evaluations (ISEs) for Phase 1 and Phase 2 of Order EA-13-109 for Monticello by letters dated April 2, 2015 (ADAMS Accession No. ML15082A167), and September 6, 2016 (ADAMS Accession No. ML16244A120), respectively. When developing the ISEs, the staff identified open items where the staff needed additional information to determine whether the licensee's plans would adequately meet the requirements of Order EA-13-109.

The NRC staff is using the audit process described in letters dated May 27, 2014 (ADAMS Accession No. ML14126A545), and August 10, 2017 (ADAMS Accession No. ML17220A328), to gain a better understanding of licensee activities as they come into compliance with the order. As part of the audit process, the staff reviewed the licensee's closeout of the ISE open items.

The NRC staff conducted a teleconference with the licensee on March 22, 2018. The enclosed audit report provides a summary of that aspect of the audit.

If you have any questions, please contact me at (301) 415-1025 or by e-mail at Rajender.Auluck@nrc.gov.

Sincerely,

RAuluck

Rajender Auluck, Senior Project Manager
Beyond-Design-Basis Engineering Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure:
Audit report

cc w/encl: Distribution via Listserv

AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AUDIT OF LICENSEE RESPONSES TO INTERIM STAFF EVALUATIONS OPEN ITEMS RELATED TO ORDER EA-13-109 MODIFYING LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS

NORTHERN STATES POWER COMPANY - MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

BACKGROUND

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 Condition," to all Boiling-Water Reactor (BWR) licensees with Mark I and Mark II primary containments. The order requirements are divided into two parts to allow for a phased approach to implementation.

Phase 1 of Order EA-13-109 requires license holders of BWRs with Mark I and Mark II primary containments to design and install a Hardened Containment Vent System (HCVS), using a vent path from the containment wetwell to remove decay heat, vent the containment atmosphere (including steam, hydrogen, carbon monoxide, non-condensable gases, aerosols, and fission products), and control containment pressure within acceptable limits. The HCVS shall be designed for those accident conditions (before and after core damage) for which containment venting is relied upon to reduce the probability of containment failure, including accident sequences that result in the loss of active containment heat removal capability or extended loss of alternating current (ac) power (ELAP). The order required all applicable licensees, by June 30, 2014, to submit to the Commission for review an overall integrated plan (OIP) that describes how compliance with the Phase 1 requirements described in Order EA-13-109 Attachment 2 will be achieved.

Phase 2 of Order EA-13-109 requires license holders of BWRs with Mark I and Mark II primary containments to design and install a system that provides venting capability from the containment 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 order required all applicable licensees, by December 31, 2015, to submit to the Commission for review an OIP that describes how compliance with the Phase 2 requirements described in Order EA-13-109 Attachment 2 will be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14183A412), Northern States Power Company - Minnesota (NSPM, the licensee) submitted its Phase 1 OIP for Monticello Nuclear Generating Plant (MNGP, Monticello). By letters dated December 16, 2014, June 22, 2015, December 17, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 17, 2016, December 19, 2016, June 14, 2017, and December 21, 2017 (ADAMS Accession Nos. ML14353A215, ML15173A176, ML15356A120, ML16169A309, ML16354A666, ML17166A051, and ML17355A508, respectively), the licensee submitted its 6-month updates to the OIP. as required by the order.

The NRC staff reviewed the information provided by the licensee and issued interim staff evaluations (ISEs) for Phase 1 and Phase 2 of Order EA-13-109 for Monticello by letters dated April 2, 2015 (ADAMS Accession No. ML15082A167), and September 6, 2016 (ADAMS Accession No. ML16244A120), respectively. When developing the ISEs, the staff identified open items where the staff needed additional information to determine whether the licensee's plans would adequately meet the requirements of Order EA-13-109.

The NRC staff is using the audit process in accordance with the letters dated May 27, 2014 (ADAMS Accession No. ML14126A545), and August 10, 2017 (ADAMS Accession No. ML17220A328), to gain a better understanding of licensee activities as they come into compliance with the order. The staff reviews submitted information, licensee documents (via ePortals), and preliminary Overall Program Documents (OPDs)/OIPs, while identifying areas where additional information is needed. As part of this process, the staff reviewed the licensee closeout of the ISE open items.

AUDIT SUMMARY

As part of the audit, the NRC staff conducted a teleconference with the licensee on March 22, 2018. The purpose of the audit teleconference was to continue the audit review and provide the NRC staff the opportunity to engage with the licensee regarding the closure of open items from the ISEs. As part of the preparation for this audit call, the staff reviewed the information and/or references noted in the OIP updates to ensure that closure of ISE open items and the HCVS design are consistent with the guidance provided in Nuclear Energy Institute (NEI) 13-02, Revision 1 and related documents (e.g. white papers (ADAMS Accession Nos. ML14126A374, ML14358A040, ML15040A038 and ML15240A072, respectively) and frequently asked questions (FAQs), (ADAMS Accession No. ML15271A148)) that were developed and reviewed as part of overall guidance development. The NRC staff audit members are listed in Table 1. Table 2 is a list of documents reviewed by the staff. Table 3 provides the status of the ISE open item closeout for Monticello. The open items are taken from the Phase 1 and Phase 2 ISEs issued on April 2, 2015, and September 6, 2016, respectively.

FOLLOW UP ACTIVITY

The staff continues to audit the licensee's information as it becomes available. The staff will issue further audit reports for Monticello, as appropriate.

Following the licensee's declarations of order compliance, the licensee will provide a final integrated plan (FIP) that describes how the order requirements are met. The NRC staff will evaluate the FIP, the resulting site-specific OPDs, as appropriate, and other licensee documents, prior to making a safety determination regarding order compliance.

CONCLUSION

This audit report documents the staff's understanding of the licensee's closeout of the ISE open items, based on the documents discussed above. The staff notes that several of these documents are still preliminary, and all documents are subject to change in accordance with the licensee's design process. In summary, the staff has no further questions on how the licensee has addressed the ISE open items, based on the preliminary information. The status of the NRC staff's review of these open items may change if the licensee changes its plans as part of final implementation. Changes in the NRC staff review will be communicated in the ongoing audit process.

Attachments:
1. Table 1 – NRC Staff Audit and Teleconference Participants
2. Table 2 – Audit Documents Reviewed
3. Table 3 – ISE Open Item Status Table

Table 1 - NRC Staff Audit and Teleconference Participants

Title	Team Member	Organization
Team Lead/Sr. Project Manager	Rajender Auluck	NRR/DLP
Project Manager Support/Technical Support – Containment / Ventilation	Brian Lee	NRR/DLP
Technical Support – Containment / Ventilation	Bruce Heida	NRR/DLP
Technical Support – Electrical	Kerby Scales	NRR/DLP
Technical Support – Balance of Plant	Garry Armstrong	NRR/DLP
Technical Support – I&C	Steve Wyman	NRR/DLP
Technical Support – Dose	John Parillo	NRR/DRA

Table 2 – Audit Documents Reviewed

Calculation 16-006, "Hard Pipe Vent D8 Battery HCVS 125VDC Battery Calculation," Revision 1
Engineering Change (EC) 23964 – FLEX 480 V Diesel Generator Sizing
Calculation 94-017, "Calculation of Alternate Nitrogen System Supply Pressure and Spare Bottle Inventory," Revision 10B
Calculation 16-011, "Calculation of HPV System Dedicated Nitrogen Supply and Pressure Requirements," Revision 0A
Calculation 16-055, "Monticello GOTHIC Analysis for the Hardened Contianment Vent Project," Revision 0
Calculation 16-054, "MNGP HCVS Radiological Assessment," Revision 0
Calculation 16-019, "Monticello Hardened Containment Vent System (HCVS) Capacity Analysis and Verification of Suppression Pool Decay Heat Capacity," Revision 0
Engineering Evaluation (EE) 26081-01 – Reasonable Protection Evaluation Grade for HCVS Tornado Missile Barrier
Calculation 16-032, "Hardened Containment Vent Pipe Supports HPVH1, HPVH2, HPVH3, and HPVH4," Revision 11
Calculation 16-012, "Pipe Stess Analysis of Hard Pipe Vent," Revision 0
Calculation 16-003, "Evaluation of HPV Missile Barrier - Lower Frame," Revision 0
Engineering Change (EC) 28557 – PT-7251B - Severe Accident Temperature Conditions
Engineering Evaluation EC 28582 – BDBEE Environmental Conditions for LT-7338B, Revision 0
Environmental Qualification (EQ) 98-039 – Rosemount Pressure Transimitter Series A (DOR), Revision 0
Environmental Qualification (EQ) 08-016 – Rosemount 1154 Transimitters, Revision 1
Engineering Evaluation EC 28546 – BDBEE Environmental Conditions for AO-4539 and AO-4540, Revision 1
Specification NPD-M-39, "Specification for Valve Requirements for Pneumatic Operated Butterfly Valves for the Hard Pipe Vent System," Revision 8
Qualification Summary Report 04518900-QSR – HCVS Radiation Monitoring System (DC & AC Input Power Supplies), Revision C
Operations Manual Section B.08.08-01, "Plant Communications Systems," Revision 7
Operations Manual Section A.8-06.02, "Repower PAB PBX Phone System with Portable Generator," Revision 3
Engineering Change (EC) 26083, "Hardened Containment Venting System NRC Order EA-13-109 Phase 1," Revision 0
Operations Manual Section C.5.-3505, " Venting Primary Containment," Revision 14
Calculation 16-002, "Evaluation of HPV Missile Barrier – Upper & Intermediate Frames," Revision 2
Calculation 16-067, "HCVS Radiation Detector Support Evaluation," Revision 0
Calculation 16-059, "Seismic Evaluation of SPOTMOS Panel C-289B," Revision 0
Calculation 16-065, "Seismic Evaluation of Panel C-292," Revision 0
Calculation 03-008, "AOV Component Calculation, Hard Pipe Vent Valves, AO-4539 and AO-4540," Revision 5
EPRI Technical Report 3002003301 – Technical Basis for Severe Accident Mitigating Strategies, Volume 1

Engineering Evaluation 28694 – Evaluation of Radiological Conditions at the Southside of the Radwaste Building during Hard Pipe Vent (HPV) Use As An Optional Location for the Portable Diesel Pump
Environemental Qualification (EQ) 98-026, "Limitorque Motor Operators (50.49)," Revision 2
Engineering Evaluation 608000000102 – SAWA Flowrates and Torus Water Levels
Calculation 16-057, "3rd Floor EFT Exhaust Fan," Revision 0
Calculation 16-022, "Ventilation Requirements for Batteries Located on the Third Floor of theft Building," Revision 0
Specifications for Model EL 2200 Electromagnetic Flow Meter
BWROG-TP-008, "Severe Accident Water Addition Timing"
BWROG-TP-011, "Severe Accident Water Management Supporting Evaluations"

Monticello Nuclear Generating Plant
Vent Order Interim Staff Evaluation Open Items:

Table 3 - ISE Open Item Status Table

ISE Open Item Number Requested Action	Licensee Response – Information provided in 6 month updates and on the ePortal	NRC Staff Close-out notes	Safety Evaluation (SE) status Closed; Pending; Open (need additional information from licensee)
Phase 1 ISE 1 Make available for NRC staff audit the final sizing evaluation for HCVS batteries/Battery charger including incorporation into. FLEX DG loading calculation.	A calculation has been performed that confirms that the HCVS battery and battery charger are sized adequately. The results of the analysis show that the battery is adequately sized to supply power to the HCVS devices for twenty-four (24) hours following the onset of an ELAP. The analysis results also show that the minimum calculated terminal voltage at the devices is above the minimum voltage required for each HCVS device while being supplied from the battery. The design allows for use of the Diverse and Flexible Coping Strategies (FLEX) equipment (i.e. FLEX generator) to power the system after 24 hours. The design incorporates a manual, break-before-make transfer switch to transfer the load from the normal HCVS power supply to 250VDC [volts direct current] battery number 16. During an ELAP event, the 16 battery, through its associated battery charger, will be connected to and powered from the FLEX portable diesel generator per procedure. An engineering evaluation was performed to demonstrate that the FLEX 480 V	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee stated that all electrical power required for operation of HCVS components is provided by the HCVS 125 VDC battery and battery charger. The battery sizing calculation 16-006, "Hard Pipe Vent D8 Battery HCVS 125VDC Battery Calculation," Revision 1 confirmed that the 125 VDC battery has a minimum capacity capable of providing power for 24 hours without recharging, and therefore is adequate. The licensee provided Engineering Change (EC) 23964 – FLEX 480 V Diesel Generator Sizing, which discusses re-powering of the HCVS 125 VDC battery charger using the FLEX DG. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

	Diesel Generator is of adequate size to support these loads. The evaluation determined that the FLEX 480 V Diesel Generator is capable of supplying the battery chargers for the 11, 12, 13, and 16 batteries at current limits. Therefore, the FLEX 480 V Diesel Generator has the required capacity to supply the HCVS loads since it is sized for the full capacity of the battery chargers. The calculation and evaluations have been provided to the NRC on the eportal.		
Phase 1 ISE OI 2 Make available for NRC staff audit documentation of the HCVS nitrogen pneumatic system design sizing and location.	A calculation has been performed that confirms that the HCVS two (2) nitrogen supply systems that provide pneumatic capacity to the HCVS rupture disc and containment isolation valves are sized adequately. This calculation determined that one (1) nitrogen bottle is required to fully burst the HCVS rupture disc and two (2) nitrogen bottles are required to actuate the primary containment isolation valves over 24 hours. Two (2) new nitrogen supply systems are installed in the 931' east Turbine Building with a remote manual operating station located south of the nitrogen bottles near the B Alternate Nitrogen supply. Pneumatic tubing was routed through the Turbine Building, Condenser Room, Reactor Core Isolation Cooling (RCIC) Room, and Torus Room to the HCVS rupture disc and containment isolation valves. The primary location for control of the HCVS remains in the third floor Emergency Filtration Train (EFT) Building at the Alternate Shutdown System (ASDS) panel.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation 94-017, "Calculation of Alternate Nitrogen System Supply Pressure and Spare Bottle Inventory," Revision 10B and Calculation 16-011, "Calculation of HPV System Dedicated Nitrogen Supply and Pressure Requirements," Revision 0A discusses the pneumatic design and sizing. For rupture disc, the licensee determined that one bottle of nitrogen can rupture the disc in 12 minutes (which is less than the required 15 minutes) to supply nitrogen upstream for HCVS operation. A spare nitrogen bottle will be stored in the Monticello warehouse on site.	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

	The design of the new HCVS nitrogen system is provided in Figure OI 2-1 of the Sixth 6-Month Status Update submittal. The calculation and drawings for the new nitrogen systems were provided to the NRC on the eportal.	For hard pipe vent (HPV) supply, the licensee determined that 2 bottles of nitrogen will be needed for 8 air operated valves (AOV) actuations for 24 hours. An additional minimum of 12 nitrogen bottles will be needed for 6 days after the initial 24 hours for more AOV actuations for the HCVS. No follow-up questions.	
Phase 1 ISE OI 3 Make available for NRC staff audit an evaluation of temperature and radiological conditions to ensure that operating personnel can safely access and operate controls and support equipment.	The primary operating station (POS) for the HCVS is in the third floor of the EFT building and includes the controls for the HCVS as well as the instruments used to monitor drywell pressure, suppression pool level, HCVS radiation, and HCVS temperature. The remote operating station (ROS) is located in the 931' elevation of the turbine building east side. The nitrogen bottle rack, controls, and indicators are located at the north end of 931' east and the ROS valves are located at the south end of 931' east. Dose rates due to the Beyond Design Basis External Event (BDBEE) and the HCVS order severe accident conditions assumed in the containment atmosphere during HPV operation were determined by calculation using the methodology in NEI-13-02, Rev 1 and HCVS-WP-02, Rev 0. The seven day integrated dose values at the POS and ROS locations are well within the dose limit of 5 rem. Transit paths and locations outside of the Reactor and/or HPCI Building have unlimited access up to 7 hours after ELAP. Additionally, transit paths are acceptable for short durations after venting has	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation 16-055, "Monticello GOTHIC Analysis for the Hardened Containment Vent Project," Revision 0 indicates that the temperature in the Emergency Filtration Train (EFT) building third floor (location of the primary operaring 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 indicates the ETF Building 3rd floor temperature varies between 110°F and 100°F with the daily diurnal temperature variation after supplemental ventilation is installed. The NRC staff requested clarification that the high temperature in the POS would not hinder operators ability	Closed [Staff evaluation to be included in SE Sections 3.1.1.2 and 3.1.1.3]

	started based on the expected peak dose rates. The FLEX Pump and FLEX Generator deployment locations were evaluated for a 7-day integrated dose and selected locations are accessible. Dose the operator receives is administratively controlled by health physics personnel to ensure set dose rates and dose limits are not exceeded. Temperature in the EFT building third floor (e.g. POS) during an ELAP in the summer will peak at approximately 135°F [degrees Farenheit] at 12 hours. By hour 12, supplemental ventilation will be installed per procedure and room temperature will then be maintained below 120°F for the duration of the 7 day period. Room temperature in the winter will drop to 35°F after 24 hours and 0°F at the end of 7 days with no mitigating actions taken. Procedures direct operators to add portable heaters as needed within 15 hours upon initiation of an ELAP to maintain EFT building third floor temperatures above 40°F. Temperature in the Turbine Building 931' east side corridor (near the ROS) in the winter will drop to 29°F after 24 hours and 0°F at the end of 7 days with no mitigating actions taken. HCVS equipment in this area can perform its function in these low temperature conditions and therefore is acceptable. Summer peak temperatures in this area are not a concern due to a lack of heat loads in the area during an ELAP.	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 peroidic monitioring 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 winter, the same procedure (Procedure C.5-4503) instructs operators to use portable heaters as needed to maintain the temperature above 40°F. The licensee concluded the summer temperature at the remote operating station (ROS) are not a concern since there are no heat loads. There is no equipment adversely affected by cold temperatures. The ROS is not continuously occupied. Operators can perform required actions independent of the local ROS temperature. Calculation 16-054, "MNGP HCVS Radiological Assessment," Revision 0 was performed to determine the integrated radiation dose due to HCVS operation. The NRC staff reviewed this calculation and determined that the licensee used conservative assumptions and followed the guidance outlined in NEI 13-02	

	The different pathways between the Reactor Building, EFT Building, and Turbine Building were analyzed and it was determined that there are no substantial heat sources in these areas that would cause a significant change in temperature. The analyses and supporting information described has been provided to the NRC in the eportal.	Rev.1 and HCVS-WP-02 Rev.0. Based on the expected integrated whole body dose equivalent in the POS and ROS and the expected integrated whole body dose equivalent for expected actions during the sustained operating period, the NRC staff believes that the order requirements are met. Temperature and radiological conditions should not inhibit operator actions needed to initiate and operate the HCVS during an ELAP with severe accident conditions. No follow-up questions.	
Phase 1 ISE OI 4 Make available for NRC staff audit analyses demonstrating that HCVS has the capacity to vent the steam/energy equivalent of one percent of licensed/rated thermal power (unless a lower value is justified), and that the suppression pool and the HCVS together are able to absorb and reject decay heat, such that following a reactor shutdown from full power containment pressure is restored and then maintained below the primary containment design pressure and the primary containment pressure limit.	A calculation has been performed that confirms that the modified HCVS configuration with the additional check valve has the capacity to vent the steam/energy equivalent of one (1) percent of the current licensed/rated thermal power of 2004 megawatt thermal (MWT) while maintaining containment pressure below design and Primary Containment Pressure Limit (PCPL). Additionally, this analysis evaluates the capacity of the Suppression Pool (SP) to absorb decay heat following a reactor shutdown from full power. The calculation has been provided to the NRC on the eportal.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation MNGP 16-019 Revision 1, "Monticello Hardened Containment Vent System (HCVS) Capacity Analysis and Verification of Suppression Pool Decay Heat Capacity," determined that 1% of the licensed thermal power (2004 MWt) venting requirement is 75,718 lbm/hr at 62 per square in gauge (psig) (PCPL = 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	Closed [Staff evaluation to be included in SE Section 3.1.2.1]

		wetwell with the torus completely full of water, is 79,737 lbm/hr (5.3% flow margin to 1% thermal power requirement). Flow varies from roughly 20,000 lbm/hr at 5 psig to 90,000 lbm/hr at 55 psig. No follow-up questions.	
Phase 1 ISE OI 5 Make available for NRC staff audit the seismic and tornado missile final design criteria for the HCVS stack.	HCVS piping outside the Class I structure is designed for tornado/wind loads without failure to ensure functionality of the HCVS and safety related systems in the vicinity. HCVS piping up to and including the second primary containment isolation valve is designed to safety related seismic Class 1 requirements. HCVS piping downstream of the second containment isolation valve, although non-safety related, is designed to seismic Class 1 as it must remain functional following a seismic event. Analysis of the tornado/wind loads and seismic loading is documented in calculations performed to support the design of the HCVS piping. The analysis of the modified HCVS piping includes incorporation of wind, tornado, and updated seismic requirements to meet sections 5.1.1.6 and 5.2 of NEI 13-02. Design basis loading requirements for wind, tornado, and seismic were used as described in the MNGP USAR [updated safety analysis report], Section 12.02. Portions of the HCVS outside of Class I structures will be protected from tornado missile impact up to 30 feet (ft) above grade. The HCVS design will meet assumptions found in guidance document	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Engineering Evaluation (EE) 26081-01 – Reasonable Protection Evaluation Grade for HCVS Tornado Missile Barrier, evaluated the HCVS stack. The licensee's HCVS design meets the assumptions found in guidance document HCVS-WP-04. No follow up questions.	Closed [Staff evaluation to be included in SE Section 3.2.2]

	HCVS-WP-04 which provides reasoning why protecting the HCVS 30 ft above grade is not required. An Engineering Evaluation validated the guidance is applicable for use at MNGP. Missile barrier design requirements for tornado generated missiles, seismic, and wind loadings were used as described in the MNGP USAR, Section 12.02. Analysis of the missile barrier to these loading requirements is documented in calculations. The calculations and analyses described above have been provided to the NRC on the eportal.		
Phase 1 ISE OI 6 Make available for NRC staff audit the descriptions of local conditions (temperature, radiation and humidity) anticipated during ELAP and severe accident for the components (valves, instrumentation, sensors, transmitters, indicators, electronics, control devices, and etc.) required for HCVS venting including confirmation that the components are capable of performing their functions during ELAP and severe accident conditions.	The POS for the HCVS is on the third floor of the EFT building and includes the controls for the HCVS as well as the instruments used to monitor drywell pressure, suppression pool level, HCVS radiation, and HCVS temperature. The ROS is located on the south end of the 931' elevation of the Turbine Building east side. The nitrogen bottle rack, controls, and pressure indicators are located at the north end of the 931' elevation of the Turbine Building east side. The primary containment isolation valves (PCIVs) and associated solenoid valves (SVs) are installed in the vent piping near the torus connection in the Reactor Building elevation 923' above the north east section of the torus. The suppression pool level transmitter LT7338B is located in the torus room bay 9.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. EC 26083 discusses the environmental conditions during an accident at the locations containing instrumentation and controls (I&C) components. The staff's review indicated that the environmental qualification met the order requirements. The primary control location is on the third floor of the EFT building. Controls for the existing HPV are located on the C-292 Alternate Shutdown System (ASDS) panel. The remote operating station is on the 931' elevation of the Turbine Building. Temperature for these areas evaluated in calc	Closed [Staff evaluation to be included in SE Section 3.1.1.4]

	The radiation detector is installed adjacent to the pipe above the high pressure coolant injection (HPCI) room at elevation 935'. The temperature element is installed in the HPCI room adjacent to the vent pipe at elevation 928'. The drywell pressure transmitter PT7251B is located in the Reactor Building, elevation 985' south wall. <u>Radiological Conditions</u>: Radiological dose rates resulting from HCVS venting were determined by calculation for each area using the methodology in NEI-13-02, Rev 1 and HCVS-WP-02, Rev 0. <u>Temperature/ Humidity Conditions</u>: Temperature conditions for each area have been determined by calculation, using the methodology in NEI-13-02, Rev 1. An additional analysis was performed to determine the severe accident temperature in the torus room. The calculations determined that key components necessary for HCVS venting are capable of performing their intended functions under ELAP and severe accident conditions. The analyses and supporting information that support these conclusions have been provided to the NRC in the eportal.	16-055. The calculation assumed a 95°F outdoor temperature. The calculation determined the ETF Bldg, 3rd floor peaks at ~135°F shortly after start of the event and drops to approximately 100°F after mitigating actions are implemented. The temperature varies between 110°F and 100°F with the daily diurnal temperature variation. The main control room was previously evaluated as part of Order EA-12-049. No follow up questions.	
Phase 1 ISE OI 7	The HCVS controls are located on the ASDS panel located on the third floor of	The NRC staff reviewed the information provided in the 6-	Closed

Make available tor NRC staff audit documentation that demonstrates adequate communication between the remote HCVS operation locations and HCVS decision makers during ELAP and severe accident conditions.	the EFT building. Primary containment pressure and suppression pool level indicators are located on the ASDS panel. Suppression pool temperature, HCVS temperature, and HCVS radiation indicators are on the panel adjoining the ASDS panel. These are the indicators used by the Operator to monitor the primary containment and HCVS when making decisions regarding use of the HCVS during severe accident conditions. When dispatched from the control room, the Operator sent to the ASDS panel will have been given a containment pressure control band by the Control Room Supervisor per procedure. Procedural guidance for operating the HCVS is maintained both in the control room and at the ASDS panel. Therefore, the Operator actuating the HCVS from the ASDS panel requires no further communication. Should actuation of the HCVS from the ASDS panel fail, the HCVS can be actuated by an Operator manipulating manual valves at the ROS, located on the east side of the 931 foot elevation of the Turbine Building. This Operator will be in communication with a second Operator who is at the ASDS panel monitoring the primary containment and HCVS. These Operators will be in communication via the telephone system. There is a phone on the ASDS panel and a phone in the Turbine Building, a short distance from the HCVS ROS.	month updates and on the ePortal. The communication methods are the same as accepted in Order EA-12-049. No follow-up questions.	[Staff evaluation to be included in SE Section 3.1.1.1]

	The MNGP phone system is powered by the Non-1E Uninterruptable Power Supply (Y91), which is powered from the site non-essential 250 volt battery. A calculation determined that the non-essential 250 volt battery will maintain power to the portion of the site phone system supplied from Y91 energized for 12 hours following an ELAP event. Phones that remain energized include the phone at the ASDS panel, the Control Room Supervisor's phone in the Main Control Room, and the phone in the Turbine Building near the HCVS ROS. In response to NRC Order EA-12-049 (Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events), NSPM developed and implemented FLEX Support Guidelines (FSGs) to provide pre-planned procedures to improve the stations capability to cope with beyond design basis events. As part of the FLEX response, MNGP has an FSG procedure to stage a 120 volt portable diesel generator and a procedure to use this generator to repower the phone system. Timing studies performed as part of FLEX implementation have shown the phone system can be repowered from the portable diesel generator within 12 hours. Since the phones required for communication at the ASDS panel and the HCVS ROS will be repowered from a portable diesel generator before power is lost from the site non-essential 250 volt battery, the phone system remains		

	available at all times for communication between the Operator at the HCVS ROS and the Operator at the ASDS panel. The calculation and procedures described in this response have been provided to the NRC on the eportal.		
Phase 1 ISE OI 8 Provide a description of the final design of the HCVS to address hydrogen detonation and deflagration.	The risk of hydrogen detonation and deflagration has been mitigated in the design of the MNGP HCVS system by use of the following elements: - A check valve will be installed on the HCVS piping at the reactor building roof to prevent ingress of air when venting stops and the steam condenses. This will prevent a flammable mixture of gasses from potentially building up within the piping upstream of the check valve. Piping downstream of the check valve will be at a length less than the recommended run up distance in order to rule out detonation loading in this portion of the piping. HCVS piping where the check valve is installed will be routed slightly over the reactor building roof to allow for maintenance/testing accessibility, then routed upwards to direct effluent away from plant structures. This is consistent with Option 5 of NEI 13-02, Appendix H. - A check valve will be utilized on the rupture disc pneumatic supply connection to the HCVS piping to prevent backflow to the remote operating station. With exception of the rupture disc supply, the HCVS piping is designed to have no interfaces with other plant systems. In addition, HCVS pneumatic system valves	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee's design is consistent with Option 5 of the NRC staff endorsed white paper HCVS-WP-03. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.11]

	that open external to the system are designed to the system operating conditions. With these design features, the HCVS meets the requirement for minimizing the potential for hydrogen gas migration and ingress into site buildings. This is consistent with the guidance provided in NEI 13-02, Section 4.1.2, Appendix H, HCVS-FAQ-05 and HCVS-WP-03. - The HCVS release point will be modified from its current location of 3 ft above the Reactor Building roof and plenum exhaust to be vertical over the reactor building roof. The modified release point is at an elevation higher than the adjacent power block structures, which is approximately 145 ft off the ground. The existing "T" type exhaust at the top of the vent pipe will be replaced with a vertical exit to direct the effluent away from site structures and away from ventilation system intake and exhaust openings. A weather cap will be installed at the release point for protection of the pipe and newly installed check valve during normal operation, and will be designed to blow off if the vent is operated. The weather cap will be designed to blow off at a minimal interior pipe pressure to not impede the initial venting. This design allows for no permanently added resistance to piping for effluent flow. This is consistent with the guidance provided in NEI 13-02 Section 4.1.5, Appendix H and HCVS-FAQ-04.		

	- With the exception of the rupture disc supply, the HCVS piping is designed to have no interfaces with other plant systems. In addition, HCVS pneumatic system valves that open external to the system are designed to the system operating conditions. With these design features, the HCVS meets the requirement of minimizing unintended cross flow within the unit. MNGP is a single unit site, so cross flow between units is not a concern. This is consistent with the guidance provided in NEI 13-02, Sections 4.1.2, 4.1.4 and 4.1.6 and HCVS-FAQ-05. The engineering change describing the above design elements has been provided to the NRC on the eportal.		
Phase 1 ISE OI 9 Provide a description of the strategies for hydrogen control that minimizes the potential for hydrogen gas migration and ingress into the reactor building or other buildings.	The HCVS utilizes a dedicated penetration from the torus to HCVS piping, which is routed through the Reactor Building. The HCVS piping does not pass through other buildings thus eliminating the potential for migration of hydrogen gas from the HCVS into other buildings. A check valve is provided on the rupture disc pneumatic supply connection to the HCVS piping to prevent backflow to the remote operating station. With exception of the rupture disc pneumatic supply, the HCVS piping is designed to have no interfaces with other plant systems, and all valves that open external to the system are designed to the system operating conditions. Once the rupture disk is burst the pneumatic supply will be isolated to	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The NRC staff's review of the proposed system indicates that the licensee's design appears to meet the requirement for minimizing the potential for hydrogen gas migration and ingress into the Reactor Building or other site buildings. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.12]

	prevent migration of hydrogen gas into the pneumatic supply system. Initial and periodic testing of the HCVS will be performed in accordance with manufacturer instructions and the NEI 13-02 guidance. This includes leak tests which will ensure leak tightness of the HCVS to prevent hydrogen gas ingress into the Reactor Building. Finally, the HCVS outlet is above plant structures, and is designed to direct the vent discharge away from structures and ventilation inlets and outlets. With these design features, the HCVS meets the requirement for minimizing the potential for hydrogen gas migration and ingress into the Reactor Building or other site buildings. The design documents and procedures described in this response have been provided to the NRC on the eportal.		
Phase 1 ISE OI 10 Make available for NRC staff audit descriptions of all instrumentation and controls (i.e., existing and planned) necessary to implement this order including qualification methods.	<u>Required Instrumentation and Controls</u>: As documented in the MNGP Overall Integrated Plan (OIP), the following instrumentation and controls are required for order compliance: • Valve Position Indication • Effluent Discharge Radioactivity • Effluent Temperature • Containment Pressure • Wetwell Level • Electrical Power • Remote Operating Station Valves	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The existing plant instuments required for HCVS (i.e. wetwell level instruments and drywell pressure instruments) meet the requirements of Regulatory Guide (RG) 1.97. The licensee provided analyses and/or supporting information of the HCVS instruments and	Closed [Staff evaluation to be included in SE Section 3.1.2.8]

	• Pneumatic Supply Pressure Indications and Manual Valves <u>Qualification Methods</u>: The OIP provides the following information related to component qualification: "The HCVS instruments, including valve position indication, process instrumentation, radiation monitoring, and support system monitoring, will be qualified by using one or more of the three methods described in the ISG, which includes: 1. Purchase of instruments and supporting components with known operating principles from manufacturers with commercial quality assurance programs (e.g., ISO9001) where the procurement specifications include the applicable seismic requirements, design requirements, and applicable testing. 2. Demonstration of seismic reliability via methods that predict performance described in IEEE 344-2004. 3. Demonstration that instrumentation is substantially similar to the design of instrumentation previously qualified." All components were determined to have acceptable qualifications to meet the HCVS order requirements.	controls (I&C), including a description of each component and the qualification method. The staff's review indicates that the I&C components are consistent with the guidance in NEI 13-02 and its qualifications meet the order requirements. No follow-up questions.	

	The analyses and supporting information that support these conclusions have been provided to the NRC in the eportal.		
Phase 1 ISE OI 11 Make available for NRC staff audit documentation of an evaluation verifying the existing containment isolation valves, relied upon for the HCVS, will open under the maximum expected differential pressure during BDBEE and severe accident wetwell venting.	A calculation was performed that determined that the HCVS primary containment isolation valves, AO-4539 and AO-4540, will open under the maximum differential pressure expected during Beyond Design Basis External Event (BDBEE) suppression pool venting with greater than 20% margin. The valves have been shown to open against a maximum expected differential pressure of 76.7 psid [per square inch differential]. The calculation has been provided to the NRC on the eportal.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The NRC staff reviewed calculation 03-088, "AOV Component Calculation, Hard Pipe Vent Valves, AO-4539 and AO-4540," which discusses the valve/actuator information for the PCIVs. The calculation determined the full opening maximum torque was 252 foot-pounds and the corresponding actuator capability at that required valve toque is 304 foot-pounds. The NRC staff verified the actuator can develop greater torque than PCIV's unseating torque. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.2.1]
Phase 2 ISE OI 1 Licensee to provide the plant specific justification for SAWA [Severe Accident Water Addition] flow capacity less than specified in the guidance in NEI 13-02, Section 4.1.1.2.	NEI 13-02 Section 4.1.1.2 provides the following guidance in determining the maximum flow capacity: - 4.1.1.2.1 Sites may use SAWA capacity at 500 GPM based on the generic analysis per reference 27. - 4.1.1.2.2 Sites may use a SAWA capacity equivalent to the site	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. SAWA provides cooling of core debris limiting the drywell temperature. SAWA permits venting containment through the wetwell vent without the necessity of having a drywell vent (see	Closed [Staff evaluation to be included in SE Section 4.1.1.3]

	specific RCIC design flow rate if less than 500 GPM (e.g., some sites have a RCIC design flow rate of 400 or 450 GPM). - 4.1.1.2.3 SAWA capacity less than specified in 4.1.1.2.1 or 4.1.1.2.2 should be supported by plant specific design (i.e., SAWA flow rate determined by scaling, a ratio of the plant thermal power rating over the reference plant power level multiplied by 500 GPM). NEI 13-02 Appendix C describes the basis for the reference plant SAWA flowrates (500 gpm initial flowrate, and then reduced to 100 gpm [gallons per minute] for remainder of the mission time). Guidance is provided for determining plant specific flow rates based on scaling, using the ratio of the specific plant thermal power to the reference plant thermal power. Additional basis for determining the reference plant SAWA flow rates is provided in Electiric Power Research Institute (EPRI) Technical Report 3002003301. The EPRI Report in turn references the State-of-the-Art Reactor Consequence Analyses (SOARCA) which provides the Peach Bottom (reference plant) specific analysis. Based on the established guidance, the MNGP plant specific flowrates are determined using the scaling method:	discussion for Phase 1 ISE 4 for wetwell vent capacity). SAWM manages the water addition into the wetwell such that the wetwell vent does not become blocked by the water level and remains operational. SAWA and SAWM industry study (The EPRI study (Technical Basis for Severe Accident Mitigating Strategies, 3002003301) assumes a 500 gpm SAWA injection flow) was based on a reference plant which has the most limiting containment heat capacity in the US fleet and therefore is conservative. NSPM used the SAWA injection flow rate for the reference plant prorated for the difference between the reactor thermal power level and the licensed reactor thermal power for Monticello. No follow-up questions.	

	Reference plant values: Rated thermal power= 3514 MWth SAWA flow= 500 gpm MNGP calculation: SAWA = 500 gpm * (2004/ 3514) = 285 gpm SAWM = 100 gpm * (2004/ 3514) = 57 gpm It should be noted that these values are different than those provided in the Phase 2 OIP. The original calculation used a reference plant thermal power of 3293 MWth, resulting is SAWA/SAWM [severe accident water management] values of 305/61 gpm. The analyses and supporting information described above were provided to the NRC in the eportal.		
Phase 2 ISE OI 2 Licensee to evaluate the SAWA equipment and controls, as well as the ingress and egress paths for the expected severe accident conditions (temperature, humidity, and radiation) for the sustained operating period.	Plant instrumentation for SAWM that is qualified to RG 1.97 or equivalent is considered qualified for the sustained operating period without further evaluation. The following plant instruments are qualified to RG 1.97: • PI-7251B (PT-7251B) Primary Containment Wide Range Pressure • LI-7338B (LT-7338B) Suppression Pool Level Passive components that do not need to change state after initially establishing SAWA flow do not require evaluation beyond the first 8 hours, at which time they are expected to be installed and ready for use to support SAWA/SAWM.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The drywell pressure and torus level indications are RG 1.97 compliant and are acceptable as qualified. Calculation 16-054, “MNGP HCVS Radiological Assessment,” Revision 0 shows that radiological conditions should not inhibit operator actions or SAWA equipment and controls needed to initiate and operate the HCVS during an ELAP with severe accident conditions.	Closed [Staff evaluation to be included in SE Sections 4.5.1.1, 4.5.1.2 and 4.5.1.3]

	The following additional equipment performing an active SAWA/SAWM function is considered: <u>SAWA/SAWM flow instrument</u> The environmental (temperature) capability of the flow instrument has been documented in ISE Open item 7. The deployment location for the flowmeter is inside the Turbine Building, east side, 931' elevation. In this location, the Turbine Building provides environmental protection from external events, and substantial radiation shielding from the HCVS vent line. Dose calculations performed determine that peak severe accident dose rate in this area is 0.186 R/hr with a 7-day integrated dose of 15.5 R. This radiation level is not expected to have any adverse effect on operation of the flowmeter. <u>SAWA/SAWM pump (Flex Pump)</u> The deployment and staging for the portable diesel pump is the same as FLEX strategies. The deployment routes and environmental operating conditions (temperature) have previously been addressed for FLEX. Planned staging locations are near the Intake Structure, Discharge Canal, or Cooling Tower Basins. Dose calculations performed determine the peak accident dose rates and integrated 7- day dose in these areas:	The NRC staff reviewed calculation 16-054, "MNGP HCVS Radiological Assessment," and determined that the licensee used conservative assumptions and followed the guidance outlined in NEI 13-02 Rev.1 and HCVS-WP-02 Rev.0. Based on the expected integrated whole body dose equivalent in the MCR and ROS and the expected integrated whole body dose equivalent for expected actions during the sustained operating period, the NRC staff believes that radiological conditions should not inhibit operator actions or SAWA equipment and controls needed to initiate and operate the HCVS during an ELAP with severe accident conditions. The temperature evaluation addressed in Phase 1 Open Item #6 bounds the SAWA/SAWM operation. For operation of equipment located outdoors, existing plant work controls remain applicable. No follow-up questions.	

	• Intake Structure- 3.1R/hr, 261 R (7-day integrated dose) • Discharge Canal- 0.15 R/hr, 122 R (7-day integrated dose) • Cooling Tower Basin (not calculated, but similar to Discharge canal) An alternate staging location for a flood event requires suction from the Condensate Storage Tanks (CST). An engineering evaluation was performed to determine dose rates in a staging location south of the Radwaste Building. This evaluation concludes that the dose rates would be similar to the FLEX Diesel Generator south location, which are negligible. These radiological conditions in the planned staging locations are not expected to affect pump operation. <u>SAWA/SAWM generator (FLEX generator)</u> Deployment and staging of the 480VAC portable diesel generator is the same as FLEX strategies. This is required to provide the power supply to the low pressure coolant injection (LPCI) valve via the LPCI swing bus. The deployment routes and environmental operating conditions (temperature) have previously been addressed for FLEX. Planned staging locations are near the Plant Administration Building (PAB) south entrance or east entrance. Dose calculations determined the peak accident dose rates and integrated		

	7- day dose in these areas: • PAB south, negligible dose rate and 7-day dose • PAB east- negligible dose rate and 7-day dose These radiological conditions are not expected to affect generator operation. <u>Ingress and Egress</u> Instrumentation (PI-7251B and LI-7338B): These instruments are located on the ASDS Panel in the EFT Building 3rd Floor. Dose calculations performed determine the peak accident dose rate in this area is 1.75mR/ hr. Access to this area will not be affected by the radiological conditions. <u>SAWA/SAWM flow instrument</u> Dose calculations determined the peak dose rate associated with the transit path to the flow instrument (Turbine Building 931' east side) is approximately 5 R/hr. Since the transit times to the area are short, ingress and egress are not expected to be impacted. <u>SAWA/SAWM pump (FLEX Pump)</u> As documented above, the radiological conditions for the deployment and staging locations are relatively low. The dose rates at the Intake Structure location could preclude access to that area; in that case, one of the alternate locations would be used. Access for operation and refueling of the pump would not be impacted by the radiological conditions.		

	<u>SAWA/SAWM generator (FLEX generator)</u> As documented above, the radiological conditions for the deployment and staging locations are negligible. Access for operation and refueling of the generator would not be impacted by the radiological conditions. [Note: The dose calculation performed does not consider radiation shine from the external radioactive plume. Station procedures will direct plant staff to monitor the radiological conditions in and around the plant during an emergency. Based on the specific site conditions, equipment locations, transport paths, and stay times would be altered as necessary to minimize personnel dose.] The analyses and supporting information described above were provided to the NRC in the eportal.		
Phase 2 ISE OI 3 Licensee to demonstrate how instrumentation and equipment being used for SAWA and supporting equipment is capable to perform for the sustained operating period under the expected temperature and radiological conditions.	<u>Equipment and Controls</u>: The following instrumentation and equipment has been evaluated for the expected temperature and radiological conditions (Reference the response to Phase 2 Open Item 2): - PI-7251B Primary Containment Wide Range Pressure - LI-7338B Suppression Pool Level - SAWA/SAWM flow instrument - SAWA/SAWM pump (FLEX pump)	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The NRC staff confirmed the PI-7251B Primary Containment Wide Range Pressure and LI-7338B Suppression Pool Level are previously qualified for R.G. 1.97 accident monitoring. The flow instrument qualification is discussed in Phase 2 Open Item #7 below.	Closed [Staff evaluation to be included in SE Sections 4.4.1.3 and 4.5.1.2]

	- SAWA/SAWM generator (FLEX generator) This equipment is capable of performing during the sustained operating period in the expected environmental conditions. One additional active component requires review, MO-2014 Residual Heat Removal (RHR) Division 1 LPCI Inboard Injection Valve. This valve would be electrically opened from the Main Control Room in order to establish the reactor pressure valve (RPV) injection path. The valve is located in the Reactor Building, 931' elevation, East Shutdown Cooling Room. The motor operated valve would be cycled within the first eight hours of the event. <u>Temperature</u>: A calculation determined environmental temperature profiles for various locations in the Reactor Building. The temperature in the East Shutdown Cooling Room is not calculated. It is conservative to assume this room is at the same temperature as the Torus room (highest value in the Reactor Building), which reaches approximately 170°F at 8 hours for the severe accident case. The Environmental Qualification (EQ) Report applicable to MO-2014 specifies a peak qualification temperature of 343°F, with test temperatures at or above 251°F for 96 hours. Based on this, there is high confidence the valve can be electrically opened in the first 8 hours of the accident.	The NRC staff reviewed calculation 16-054, "MNGP HCVS Radiological Assessment," and determined that the licensee used conservative assumptions and followed the guidance outlined in NEI 13-02 Rev.1 and HCVS-WP-02 Rev.0. Based on the expected integrated whole body dose equivalent in the MCR and ROS and the expected integrated whole body dose equivalent for expected actions during the sustained operating period, the NRC staff believes that the order requirements are met. No follow-up questions.	

	Radiation: A dose rate calculation determined dose rates and total 7-day integrated dose for various locations, including the Reactor Building. The dose rates in the East Shutdown Cooling Room were not calculated. It is conservative to assume this room has the same radiological conditions as the Torus room, which is the compartment below this area (does not account for any shielding effect from 931' floor slab). The peak dose rate in the Torus room (near CV4539/ CV4540) is 2.7E5 R/hr. The 7-day integrated dose is 1.14E7 R. The environmental qualification (EQ) report applicable to MO-2014 specifies a demonstrated total equivalent gamma dose of 2.04E8 Rad. Assuming that 1Rem = 1Rad for this case, the qualified dose exceeds the calculated accident dose. Based on this, there is high confidence the valve can be electrically opened in the first 8 hours of the accident. The analyses and supporting information described above were provided to the NRC in the eportal.		
Phaes 2 ISE OI 4 Licensee to demonstrate that containment failure as a result of overpressure can be prevented without a drywell vent during severe accident conditions.	The SAWA/SAWM strategy requires demonstration that the wetwell vent will remain available for the 7- day mission time (i.e. water level does not rise above the elevation of the vent connection on the torus). An Engineering Evaluation has been performed to determine wetwell water level during the event. The evaluation determines the SAWA and	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. BWROG-TP-15-008 demonstrates adding water to the reactor vessel within 8-hours of the onset of the event will limit the	Closed [Staff evaluation to be included in SE Section 4.2]

	SAWM flowrates; the RPV injection rate is specified as 285 gpm for four hours, then 57 gpm for the remainder of the 7 days. The resulting wetwell water level at 7 days is approximately 24.2 feet (elevation 922.95 feet), which is below the wetwell vent elevation of 925.21 feet (upper limit on water level instrument is 925 feet). The analysis is conservative since no mass loss through the HPV is credited. Based on this analysis, the wetwell vent capability is maintained for a 7- day mission time. The wetwell vent has been designed and installed to meet NEI 13-02 Rev 1 guidance, which ensures that it is adequately sized to prevent containment overpressure under severe accident conditions. The SAWM strategy will ensure that the wetwell vent remains functional for the period of sustained operation. MNGP will follow the guidance (flow rate and timing) for SAWA/SAWM described in BWROG-TP-15-008 and BWROG-TP- 15-011. The wetwell vent will be opened prior to exceeding the PCPL value of 62 PSIG. Therefore, containment over pressurization is prevented without the need for a drywell vent. The analyses and supporting information described above were provided to the NRC in the eportal.	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. 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. As noted under Phase 1 open item #4, the vent is sized to pass a minimum steam flow equivalent to 1% rated core power. This is sufficient permit venting to maintain containment below the lower of PCPL or of design pressure. No follow-up questions.	
Phase 2 ISE OI 5 Licensee to demonstrate how the plant is bounded by the reference plant analysis that	NEI 13-02 Appendix C provides a description of the Severe Accident Water Management strategy, and recognizes insights gained from EPRI Technical Report 3002003301.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal.	Closed [Staff evaluation to be included in SE Section 4.2.1.1]

SUBJECT: 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

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