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

January 2, 2020

Mr. Bradley J. Sawatzke
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Richland, WA 99352

SUBJECT: COLUMBIA GENERATION STATION – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NO. MF4383; EPID NO. L-2014-JLD-0045)

Dear Mr. Sawatzke:

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. ML14191A688), Energy Northwest (the licensee), submitted its Phase 1 OIP for Columbia Generation Station (Columbia) 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 Columbia, including the combined Phase 1 and Phase 2 OIP in its letter dated December 16, 2015 (ADAMS Accession No. ML15351A363). 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 March 25, 2015 (Phase 1) (ADAMS Accession No. ML14335A158), September 29, 2016 (Phase 2) (ADAMS Accession No. ML16266A233), and August 6, 2018 (ADAMS Accession No. ML18215A204), the NRC issued Interim Staff Evaluations and an audit report, respectively, on the licensee's progress. By letter dated August 20, 2019 (ADAMS Accession No. ML19232A448), the licensee reported that Columbia is in full compliance with the requirements of Order EA-13-109, and submitted a Final Integrated Plan for Columbia.

The enclosed safety evaluation provides the results of the NRC staff's review of Columbia's hardened containment vent design and water management strategy for Columbia. The intent of the safety evaluation is to inform Columbia 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, Integrated Program Management and BDB Branch, at 301-415-1025, or by e-mail at Rajender.Auluck@nrc.gov or Mr. Stephen Philpott, Project Manager, Integrated Program Management and BDB Branch, at 301-415-2365, or by email at Stephen.Philpott@nrc.gov.

Sincerely,

/RA/

David J. Wrona, Chief
Integrated Program Management and BDB Branch
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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RELATED TO ORDER EA-13-109

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

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], Energy Northwest (the licensee), submitted its Phase 1 Overall Integrated Plan (OIP) for Columbia Generating Station (Columbia) in response to Order EA-13-109. By letters dated December 17, 2014 [Reference 3], June 30, 2015 [Reference 4], December 16, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 30, 2016 [Reference 6], December 29, 2016 [Reference 7], June 27, 2017 [Reference 8], December 28, 2017 [Reference 9], June 21, 2018 [Reference 10], December 13, 2018 [Reference 11], and June 24, 2019 [Reference 12] the licensee submitted 6-month updates to its OIP. By letters dated May 27, 2014 [Reference 13], and August 10, 2017 [Reference 14], 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 15]. By letters dated March 25, 2015 (Phase 1) [Reference 16], September 29, 2016 (Phase 2) [Reference 17], and August 16, 2018 [Reference 18], the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated August 20, 2019 [Reference 19], 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 20]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012 [Reference 21], 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 22], the NRC staff issued Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents" [Reference 23], 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 24]. In the SRM for SECY-12-0157 [Reference 25], 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 26], 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 27], 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 28], 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 29], 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

Columbia is a single unit site with a General Electric BWR with a Mark II primary containment system. Containment integrity is maintained by controlling containment pressure using the HCVS. The HCVS is initiated using manual action from the main control room (MCR) 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 MCR 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 hazards identified in NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 2 [Reference 30], which 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. The FLEX diesel generator (DG) is connected in time to support the mitigation response (now it is before the HCV is opened) to support recharging the station batteries for monitoring the containment parameters. The FLEX generator will also supply the HCVS battery charger. These actions provide electric controls and motive power for sustained operation of the HCVS for a minimum of 7 days.

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 after initial system line up in the ROS, primary control of the HCVS is accomplished from the MCR. Alternate control of the HCVS is accomplished from the ROS in the DG building on the 441-foot level. FLEX actions that maintain the MCR habitable were implemented in response to NRC Order EA-12-049 [Reference 37]. Calculation TM-2195, "Remote Operating Station Access and Habitability during HCV Operation," evaluated these locations under severe accident conditions and found both accessible and habitable. These include implementing the actions in Plant Procedures Manual (PPM) 5.6.2, "Station Blackout (SBO) and Extended Loss of AC Power (ELAP) Attachments."

Habitability of the ROS does not require special actions. The ROS is in the DG building. During an ELAP, the diesel generators are not operating so there is no heat load in the building. The ROS is accessible from a pathway outside the reactor building which has been evaluated for habitability and radiological conditions to ensure operating personnel can safely access and operate the controls. The ROS is not expected to be continually manned.

The NRC staff reviewed the information provided and audited the calculations. Calculation TM-2195 predicted that the MCR maximum temperature reaches 117.2 degrees Fahrenheit (°F) (implementing actions in FLEX document TM-2187) and remains below the licensee-controlled specification limit of 120°F. The peak temperature in the ROS was determined to be 104°F. The compensatory actions in procedure PPM 5.6.2 were audited in the evaluation of the licensee's response to Order EA-12-049. The licensee uses a toolbox approach where mitigating actions are taken within the skill of the craft based on symptoms to address radiological and thermal conditions. Staff also noted that in areas with elevated temperatures, operator actions are not strenuous and stay times are limited.

Licensee calculation ME-02-14-04, "Reactor Building Accessibility and Equipment Operability During an Extended Loss of AC Power (ELAP)," Revision 2, determined environmental conditions in various rooms and determined allowable stay times based on wet globe temperature. Compensatory actions were evaluated for various scenarios. The peak ambient temperature used was 105°F. Appendix H of calculation ME-02-14-04 discusses equipment operability under predicted conditions. Appendix I of calculation ME-02-14-04 provides reactor building instrumentation locations and directions for direct readings with the Fluke meter.

Based on the evaluations, the NRC staff agrees that the temperature should not inhibit operator actions needed to initiate and operate the HCVS during an ELAP with severe accident conditions.

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 ME-02-17-12, "Habitability and Access with Radiological Conditions Expected During an Extended Loss of AC Power with Severe Accident Conditions," documents the dose assessment for designated areas inside the Columbia reactor building (outside of containment) and outside the Columbia reactor building caused by the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. Calculation ME-02-17-12 was performed using NRC-endorsed HCVS-WP-02 [Reference 31] and HCVS-FAQ-12 [Reference 32] 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 MCR which is the primary control location, and the ROS.

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[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 on 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/radialion/health-effects/rad-exposure-cancer.html

In its FIP, the licensee states that the ROS location and the travel path to the 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 the ROS to be an acceptable location for alternate control. The evaluation (as documented in calculation ME-02-17-12) 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 MCR and that under the postulated scenarios of Order EA-13-109, the MCR is adequately protected from excessive radiation dose and no further evaluation of its use is required (HCVS-FAQ-06). Alternate control of the HCVS is accomplished from the ROS in the DG building. 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 the 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 selected environmental qualification information in Table 1 of the FIP through an audit review of Columbia calculation ME-02-17-14, "Functionality of Equipment and Instrumentation During an Extended Loss of AC Power (ELAP)."

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, unintended cross flow of vented fluids, control panel, manual operation, power and pneumatic supply sources, inadvertent actuation prevention, HCVS monitoring, monitoring of effluent discharge, equipment operability, hydrogen control, and HCVS operation/testing/inspection/maintenance. 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 developed calculation ME-02-13-03, "Pipe Size and Pressure Drop Calculation for the Hardened Containment Vent System," Revision 2 providing verification of 1 percent power flow capacity at the containment design pressure (45 pounds per square inch gauge (psig)), which is lower than the primary containment pressure limit (PCPL) (121 psig). The calculation used RELAP5/SCDAPSIM, Version 3.4, to determine the maximum amount of steady state flow that could be passed from primary containment through the HCVS to the atmosphere. The driving pressures modeled were 45 psig and 14.5 psig. The 45 psig pressure corresponds to the maximum internal design pressure of containment from the primary containment vessel design specification and the 14.5 psig pressure corresponds to the approximate maximum wetwell temperature of 240°F.

The RELAP5 input was developed based on station isometric drawings and was split into seven pipes and three valves. The size, length, orientations, and loss coefficients were used as the RELAP5 input. A transient case was then run until steady state conditions were reached. For the 45 psig case, RELAP5 directly outputs the steady state mass flow rate. For the 14.5 psig case, the vent piping loss coefficient was determined to assure it was consistent with the limitation expressed in calculation ME-02-14-13. The loss coefficient limitation ($K \leq 4.6$) is needed to assure the suppression pool temperature remains below 240ºF (required for long-term reactor core isolation cooling operability). The mass flow, velocity, density, and pressure of the fluid at different positions in the pipe were outputs by RELAP5. A manual calculation was then performed on these values to calculate the overall loss coefficient, which was determined to be $K = 3.36$. The flow rate required to dissipate 1 percent of decay heat at 45 psig primary containment pressure is 132,514 pounds (mass) per hour (lbm/hr). The calculated flow rate at 45 psig primary containment pressure is 250,970 lbm/hr.

The NRC staff reviewed the information provided and audited the calculations. The NRC staff noted that a reactor thermal power of 4079 mega-watts thermal (MWt) was used, which is conservative relative to the licensed reactor thermal power of 3544 MWt. The flow rates were determined for a primary containment pressure of 45 psig. A 25 percent margin was included in the calculation. Final Safety Analysis Report (FSAR) Table 6.2-1 shows a maximum drywell pressure of 45 psig and 340°F. The suppression chamber design parameters are 45 psig and 275°F.

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 wetwell vent exits the suppression chamber from a spare penetration (X-58). Two air (pneumatically)-operated valves (AOV) that are air-to-open and spring-to-shut (normally closed, fail closed valves) are installed as primary containment isolation valves (PCIVs). These valves are PCIVs and categorized as locked closed valves. The HCVS discharge pipe is routed through the abandoned stairwell in the southeast corner of the reactor building and exits approximately at the 513-foot elevation. The discharge pipe then runs up the outside south wall of the reactor building to a release point greater than 3 feet above the parapet wall resulting in a discharge point at an approximate elevation of 677 feet. The HCVS path release point is independent of the reactor building elevated release path and release point. All effluents are exhausted above the reactor building. This discharge point was extended approximately 3 feet above the unit's reactor building parapet wall such that the release point will vent away from emergency ventilation system intake and exhaust openings, MCR location, location of HCVS portable equipment, access routes required following an ELAP and BDBEE, and emergency response facilities.

Part of the HCVS-FAQ-04 clarifies guidance on the placement of release point to ensure that vented fluids are not drawn immediately back into any emergency vent intakes. A zone of influence is defined such that for every 1 foot of vertical separation, the acceptable zone includes all points within a circle centered on the release point extending 5 feet horizontally. At Columbia, the elevation difference is about 230 feet from the HCVS release point to the MCR intake elevation. Based on the guidance, the intake needs to be within a horizontal distance of 1150 feet of the release point. The Columbia MCR remote intakes are within that distance from the HCVS. Therefore, the elevation of the release point is sufficient to preclude radiation issues from the vent effluent. The MCR emergency ventilation is not powered. As a result, no intake from the outside will occur until repowered. Therefore, the vent pipe is appropriately placed relative to this air intake.

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, "Missile Evaluation for HCVS Components 30 Feet Above Grade," Revision 0 [Reference 33], 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 in EC 13094 the HCVS vent pipe protection from external events with respect to wind-borne missiles against the supplemental guidance contained in HCVS-WP-04. This evaluation demonstrated that the pipe was robust with respect to external missiles per HCVS-WP-04 as described below.

In FSAR Section 2.1.1.1 it states that the site is located at 46° 28' 18" North Latitude and 119° 19' 58" West Longitude. In NEI 12-06, Figures 7-1 and 7-2 indicate that the site is not in a region where the wind speed is expected to exceed 130 mph. Therefore, the plant screens out for an assessment for high winds (hurricanes and tornados) for the protection and deployment of FLEX equipment, including missiles produced by these events. This also applies to the design of the external portions of the HCVS.

As stated above, the HCVS exhaust pipe exits at the approximate elevation of 513 feet. Grade elevation at the reactor building is approximately 441 feet. Therefore, the exposed HCVS exhaust pipe is greater than 30 feet above ground level which reduces the potential exposure to wind generated missile strike damage. Although the HCVS screens out for tornado loading concerns from a BDBEE, the system was designed to satisfy Columbia's design basis requirements for tornado missile loading to prevent any impact on adjacent safety related structures, systems, or components, such as the DG building. Specifically, the HCVS vent piping and associated pipe supports are designed to Seismic Category I requirements as documented in the supporting calculations in EC 13094.

The exterior vent piping, from the secondary containment penetration to the exhaust point is 16-inch Schedule 40 seamless ASTM A106 Grade B piping and is Seismic Category 1 to ensure that it will not fail in a seismic event and impact the safety-related DG building below. In calculation ME-02-14-16, the licensee evaluated the HCVS outdoor piping from the secondary containment anchor (HCV-4) to the vent exhaust. Based on the above description of the vent pipe design, the Columbia HCVS vent pipe design appears to meet the order requirement to be robust with respect to all external hazards.

Furthermore, Energy Northwest stated in its FIP for Order EA-12-049, “Energy Northwest's Notification of Full Compliance with Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis External Events"” (Agencywide Documents Access and Management System ADAMS Accession No. ML17229B506), and acknowledged in the NRC safety evaluation, “Safety Evaluation Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051” (Reference 37), that Columbia is not susceptible to high-wind hazards.

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.

Columbia is a single unit station. In its FIP, the licensee states that the HCVS was designed as a stand-alone system and is independent of other plant systems. Therefore, while the HCVS shares some robust, Seismic Category I structures with other systems, it has no mechanical or electrical interfaces with other systems. The NRC staff audited the information provided and concurs that the HCVS appears not to interface with other plant systems.

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 following the lineup of HCVS pneumatics, the HCVS will be initiated and then operated and monitored from a control panel located in the MCR. 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 once the HCVS pneumatics are lined up, the HCVS will be initiated and then operated and monitored from a control panel located in the MCR. In the event operation from the MCR is not available, the PCIVs can be manipulated from the ROS. The ROS is located in Room D113 on the 441-foot elevation of the DG building in a readily accessible location and provides a means to manually operate the wetwell vent. The ROS has two functions, initial manual lineup of the pneumatics that allows remote operation of the PCIVs from the MCR and manual bypassing of the solenoid pilot valves (SPV) allowing operation of the PCIVs from the ROS. This provides a diverse method of valve operation therefore improving 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 thermal and radiological evaluation of all the operator actions that may be required to support HCVS operation during a loss of ac power and severe accident and demonstrates 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 shows a sketch for the HCVS site layout. 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.

<u>Pneumatic Sources Analysis</u>

For the first 24 hours following the event, the newly installed standalone nitrogen system will provide the pneumatic force to burst the rupture disks and operate the hardened wetwell vent valves. The nitrogen system bottle racks are located at the ROS in the DG building. A total of 4 nitrogen bottles have been installed to the nitrogen system and can provide motive force for up to 8 venting cycles, which includes opening the vent valves (HCV-V-1 and HCV-V-2) for 24 hours. Additional bottles are available in the railroad bay and the FLEX compressors can provide the motive force after the initial 24-hour duration up to 7 days.

The licensee determined the required pneumatic supply storage volume and supply pressure set point required to operate the HCVS AOVs for 24 hours following a loss of normal pneumatic supplies during an ELAP in calculation ME-02-15-08, "Evaluation of the Hardened Containment Vent (HCV) Compressed Nitrogen Supply System for HCV-RD-60, HCV-54, HCV-AO-1&2," Revision 1. The licensee's calculation determined that the 3-bottle nitrogen rack system will provide sufficient motive force to open both valves once, and then cycle one valve up to 18 times. However, the licensee installed 4 bottles total. The licensee's calculation also determined that a minimum cylinder pressure of 2400 psig can provide sufficient capacity for operation of the HCVS valves for 24 hours following an ELAP and the safety relief valves the required number of times per the design basis.

The NRC staff audited the calculation and confirmed that there should be sufficient pneumatic supply available to provide motive force to operate the HCVS AOVs for 24 hours and the subsequent 7-day period 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, Columbia would rely on a new dedicated battery and battery charger with sufficient capacity to supply HCVS loads. The 24-volt (V) direct current (dc) HCVS battery is in the Division 2 battery room. The HCVS battery charger and distribution panel are in RPS room 2. Both rooms are in the radwaste building. The HCVS batteries and battery charger are installed where they are protected from applicable hazards. Enersys manufactured the HCVS battery.

The HCVS battery is model Enersys PowerSafe 2GN-15 with a nominal capacity of 1260 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 an evaluation for the HCVS battery/battery charger sizing requirements including incorporation into the FLEX DG loading calculation.

The NRC staff audited licensee calculation E/I-02-13-03, "Battery Sizing Calculation for the Hardened Containment Vent (HCV) System," Revision 1, which verified the capability of the HCVS battery to supply power to the required loads during the first phase of the Columbia venting strategy for an ELAP. The HCVS battery was sized in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 485-2010, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," which is endorsed by Regulatory Guide (RG) 1.212, "Sizing of Large Lead-Acid Storage Batteries," published 2015. The licensee's calculation identified the required loads and their associated ratings (current, power, and minimum system operating voltage). The licensee's battery sizing calculation also showed that minimum calculated terminal voltage at each device is above the minimum voltage required during the 24-hour discharge cycle. Therefore, the Columbia HCVS battery should have sufficient capacity to supply power for at least 24 hours.

The licensee's strategy includes repowering the HCVS battery charger within 24 hours after initiation of an ELAP. The licensee's strategy relies on one 400-kilowatt (kW) 480 Volt alternating current (Vac) FLEX DG. The licensee has a total of two 400 kW, 480 Vac FLEX DGs (one stationary and one portable). Only one of the FLEX DGs is required for the HCVS electrical strategy. 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 audited licensee calculation E/I-02-91-03, "Calculation for Division 1 and 2 and 3 Diesel Generator Loading," Revision 21. The calculation shows that the loading for the 400 kW 480 Vac FLEX DG has a margin of 114 kW which includes the HCVS battery charger. Based on the available margin and the NRC staff audit of calculation E/I-02-91-03, it appears that the FLEX DG should have sufficient capacity and capability to supply the necessary HCVS loads during an ELAP event.

Electrical Connection Points

The licensee's strategy to supply power to HCVS components requires using a combination of permanently installed and portable components. Staging and connecting the 400 kW FLEX DG were addressed under Order EA-12-049. Licensee procedure ABN-FSG-003, "DG4 Crosstie to E-MC-7A and E-MC-8A," Major/Minor Revision 6/1, and ABN-FSG-004, "DG5 Crosstie to E-MC-7A and E-MC-8A," Major/Minor Revision 6/1, provides guidance for connecting the HCVS battery charger to the 480 Vac FLEX DGs.

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 states that emergency operating procedures 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. The HCVs PCIVs must be opened to permit vent flow. The physical features that prevent an inadvertent actuation are the key lock switch for the PCIVs in the MCR and lock-closed nitrogen supply valve in the ROS isolating the pneumatics for PCIV operation. The NRC staff's audit of the HCVS 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 valve position; vent pipe temperature; effluent radiation; HCVS battery voltage; HCVS battery current, drywell pressure, and wetwell level. The NRC staff notes that drywell pressure and wetwell level are declared Columbia 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 that supports monitoring of HCVS effluent: HCVS valve position; HCVS effluent temperature; and HCVS vent line 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 of its FIP, the licensee stated that the HCVS radiation monitoring system consists of an ion chamber detector coupled to a process and control module is fully qualified for the expected environment at the vent pipe during accident conditions, and the process and control module is qualified for the mild environment in the ROS. An alternate indicator is qualified for the mild environment in the MCR. The NRC staff reviewed 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.

<u>Environmental</u>

The FLEX diesel-driven SAWA pumps and FLEX DGs will be staged outside so they will not be adversely impacted by a loss of ventilation.

As discussed above in Section 3.1.1.2, the alternate control of the HCVS is accomplished from the ROS. The licensee determined the peak temperature at the ROS is 104°F. All actions at the ROS are manual and do not require any electrical power.
The HCVS battery is in the Division 2 battery room. The HCVS battery charger, distribution panel, and supporting equipment are in the RPS Room 2. The NRC staff audited licensee calculation ME-02-17-09, "RadWaste Building and Remote Operating Station (ROS) Habitability

and Equipment Operability During Extended Loss of AC Power (ELAP)," Revision 1, which predicts the temperature profile in the radwaste building during an ELAP. The licensee determined that temperature profile at 72 hours in the Division 2 battery room and RPS Room 2 will be 108°F and 123°F, respectively.

Based on the above, the NRC staff concurs with the licensee's calculation that show the rooms located in the radwaste building will remain below the maximum temperature limit (122°F) of the HCVS batteries and maximum temperature limit (158°F) of the HCVS battery charger. 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 ME-02-17-12, "Habitability and Access with Radiological Conditions Expected During an Extended Loss of AC Power with Severe Accident Conditions," 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 ME-02-17-12 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's 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 3.1.2.3, the HCVS was designed as a stand-alone system and is independent of other plant systems. Therefore, while the HCVS shares some robust, Seismic Category I structures with other systems, it has no mechanical or electrical interfaces with other systems.

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 Columbia 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 upstream of and including the second containment isolation barrier is designed consistent with the design basis of the plant. These items include piping, piping supports, containment isolation valves, and the containment isolation valve actuators. The containment isolation valve position indication components are non-safety related but meet the requirements for environmental and seismic qualification of safety related components. The hardened vent piping between the wetwell and the reactor building roof is designed to 150 psig and 350°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 HCVS 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.

The licensee further provided Table 1 in the FIP, which contains a list of components, controls and instruments required to operate HCVS, their qualification and evaluation against the expected 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 this table 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.

The HCVS components downstream of the outboard containment isolation valve are routed in seismically qualified structures or supported from seismically qualified structures and that the HCVS does not interface with other plant systems.

In its FIP, the licensee stated that the HCVS components downstream of the outboard containment isolation valve are routed in seismically qualified structures or supported from seismically qualified structures. The HCVS does not interface with other plant systems. The HCVS 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. The exterior vent piping, from the secondary containment penetration to the exhaust point is 16-inch Schedule 40 seamless ASTM A106 Grade B piping and is Seismic Category 1 to ensure that it will not fail in a seismic event and impact the safety related DG building below.

In FIP Enclosure 2, Attachment 7, Table 1, “List of HCVS Component, Control, and Instrument Qualifications,” contains a list of components, controls, and instruments required to operate the HCVS, their qualification and evaluation against the expected conditions. All instruments are fully qualified for the expected seismic conditions so that they will remain functional following a seismic event.

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. Columbia 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 one of the portable, diesel-driven FLEX pumps to provide SAWA flow. The SAWA flow path is the same as the FLEX injection flow path and the connection will be made at the residual heat removal (RHR) V-63A low pressure injection (LPCI). The water source will be either of the spray ponds. The SAWA flow path hose routing is from the FLEX pump to into the RPV. Most of the SAWA actions take place outside the reactor building or are in locations shielded from the severe accident radiation by the thick concrete walls of the reactor building. Those that are inside the building can be accomplished prior to radiological dose conditions being unacceptable. Once SAWA flow is initiated, operators will have to monitor and maintain SAWA flow and ensure refueling of the diesel-driven equipment as necessary. Operators may also have to reduce flow as part of the severe accident water management (SAWM) strategy, if necessary.

4.1.1 Staff Evaluation

4.1.1.1 Flow Path

The SAWA flow path is the same as the FLEX primary injection flow path except the hoses routed inside the reactor building will be routed to RHR-V-63A. The SAWA flow path's primary water source are the spray ponds. Hoses are routed across the yard and into the reactor building railroad bay and up the southwest stairwell and connected to the RHR system. A flow element and manual control valve are installed at the 522' elevation of the Reactor Building (RB). Backflow prevention is provided by existing, safety-related, backflow prevention devices installed in the RHR LPCI injection line, which are leak tested using the existing leakage testing programs. Drywell pressure and wetwell level will be monitored and flow rate will be adjusted by use of throttle valves in the flow path.

4.1.1.2 SAWA Pump

In its FIP, the licensee states that the strategy is to use one of three diesel driven pumps for FLEX and SAWA. The licensee described the hydraulic analysis performed to demonstrate the capability of FLEX pumps to provide the required 450 gallons per minute (gpm) of SAWA flow. Section IV.C.9.1 of the FIP states that the FLEX pumps are protected from all applicable external hazards.

The NRC staff audited calculation ME-02-12-06, "Evaluation of the Use of Portable Equipment During an Extended Station Blackout," Revision 5, which determined the two pumps would be able to provide the required SAWA flow rate of 450 gpm. The NRC staff audited the flow rates and pressures evaluated in the hydraulic analysis and confirmed that the equipment is capable of providing the needed flow. Based on the NRC staff's audit of the FLEX pumping capabilities at Columbia, as described in the above hydraulic analysis and the FIP, it appears that the licensee has demonstrated that the FLEX pump should perform as intended to support SAWA.

4.1.1.3 SAWA Analysis of Flow Rates and Timing

The licensee developed the overall accident management plan for Columbia 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 implementing procedures are integrated into the Columbia severe accident management guidelines (SAMGs). The EPG/SAG Revision 3, implemented with emergency procedures committee Generic Issue 1314, allows throttling of SAWA 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 changed the Columbia mitigation response to fit the severe accident scenario by moving the water addition steps to earlier in the event timeline. This allows Columbia to have a single response for any BDBEE, even if it degrades to a severe accident. The licensee used NEI 12-06, Appendix E to revalidated Columbia's response. Validation Plan No. 13 shows that the SAWA pump can be deployed and commence injection prior to the HCV being opened. The studies referenced in NEI 13-02 demonstrated that establishing flow within 8 hours will protect containment. The initial SAWA flow rate will be at least 450 gpm. After a period, estimated to be about 4 hours, in 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 wetwell level. Guidance document NEI 13-02 generic analysis demonstrated that, SAWA flow could be reduced to 90 gpm after 4 hours of initial SAWA flow rate and containment would be 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 wetwell 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 121 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.

Licensee calculation ME-02-14-02, General Technical Support for Fukushima Related Licensing Documents," Revision 2 determined that the initial Columbia SAWA flowrate is 450 gpm, which is the site-specific flow rate when the site's rated thermal power is compared to the reference power level of NEI 13-02. The license reactor thermal power limit for Columbia is 3544 MWt. The initial SAWA flow (450 gpm) will be injecting to the RPV within 8 hours of the loss of injection. The reference plant flow rate is 500 gpm with a reactor reference power level of 3951

MWt, equivalent to the reference plant rated thermal power level used in NUREG-1935, State of the Art Reactor Consequence Analysis (SOARCA).

The NRC staff noted that 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 discussion in Section 3.1.2.1 for wetwell vent capacity). The SAWM manages the water addition into the wetwell such that the wetwell vent does not become blocked by the water level and remains operational. The 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. The SAWA flow for Columbia was determined by using the ratio of the plant licensed reactor thermal power over the reference plant reactor thermal power. This is consistent with the guidance in Section 4.1.1.2 of NEI 13-02.

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 Columbia is to implement the containment venting strategy utilizing SAWA and SAWM. This strategy consists of the use of the Phase 1 wetwell vent and SAWA hardware to implement a water management strategy that will preserve the wetwell 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 pump along with instrumentation and procedures to ensure that the wetwell vent is not submerged. Water from the FLEX pump takes suction from the service water spray ponds, discharging to a fire hose which traverses across the yard area and enters the reactor building at the reactor building railroad bay and up the southwest reactor building stairwell. The discharge hose is routed in the stairwell to the RHR piping at valve RHR-V-63A. 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 analysis to protect containment without requiring a drywell vent for at least 7 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 466' 2.75" (normal suppression pool level) and 491' (HCVS penetration centerline) in the wetwell provides approximately 815,900 gallons of water volume before the water level reaches the bottom of the vent pipe. A diagram of the available freeboard is shown on Attachment 1 to the FIP.

Generic assessment BWROG-TP-15-011, 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. Columbia performed calculation ME-02-14-02 demonstrating that the wetwell level will not reach the wetwell vent for at least at least 7 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 above, the SAWA flow is based on calculation ME-02-14-02 and BWROG-TP-15-011 to demonstrate that throttling SAWA flow after containment parameters have stabilized, in conjunction with venting containment through the wetwell vent will result in a stable or slowly rising wetwell 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 7 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 Conclusions

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 SAWA/SAWM Motive Force

4.3.1 Staff Evaluation

4.3.1.1 SAWA Pump Power Source

As described in Section 4.1, the licensee plans to use one of the two portable pumps to provide SAWA flow. The pumps are diesel-driven by an engine. Operators will refuel the pump and DGs using diesel fuel tank FLEX-TK-2 because of high dose rates near the underground diesel fuel oil (DFO) tanks. In its FIP, the licensee states that refueling will be accomplished initially using FLEX-TK-2 and then subsequent refueling will be conducted using the DFO tanks as the licensee stated the dose rates at the underground tanks are expected to be significantly reduced. The licensee will also provide dose maps to operators in procedures used for refueling. The pumps will be refueled by the FLEX refueling equipment that has been qualified for long-term refueling operations per Order EA-12-049.

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 drywell pressure and wetwell level are used for HCVS venting operation. These instruments are powered by the Class 1E 125 volts direct current (Vdc) and 250 Vdc Division 1 station batteries until the FLEX DG is deployed and available. The SAWA flow meters are mechanical and do not require any electrical power.

The NRC staff audited licensee dc coping calculation E/I-02-18-01, "ELAP Battery Sizing for 250V and 125V DC Division I Systems," Revision 1, which verified the capability of the Class 1E 125 Vdc and 250 Vdc station batteries to supply power to the required loads (e.g., drywell pressure and wetwell level) during the first phase of the Columbia FLEX mitigation strategy plan for an ELAP event. The NRC staff also audited licensee calculation E/I-02-91-03, which verified that the 400 kW FLEX DG is adequate to support the SAWA/SAWM electrical loads. The NRC staff confirmed that the Class 1E 125 Vdc and 250 Vdc station batteries, and 400 kW FLEX DG 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 SAWA/SAWM Instrumentation

4.4.1 Staff Evaluation

4.4.1.1 SAWA/SAWM Instruments

In Section IV.C.10 of its FIP the licensee stated, in part, that the instrumentation needed to implement the SAWA/SAWM strategy are wetwell level, drywell pressure and SAWA flow meter. The NRC staff found that drywell pressure and wetwell level are existing RG 1.97 instruments that were designed and qualified for severe accident conditions. The licensee also stated, in Table 1 of its FIP, that SAWA flow instrument range is 50 to 600 gpm. The SAWA flow instrument range appears to be consistent with the licensee's strategy. The NRC staff reviewed the FIP including Section IV.C.10.1, Section IV.C.10.2, Section IV.C.10.3, and Table 1 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 Columbia strategy may also use 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 a propeller style flow meter located in the RB 522-foot elevation when used. There is no electrical power needed as it is totally mechanical.

The NRC staff reviewed the FIP, including Table I and Section IV.C.10.2 and found 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

Drywell pressure and wetwell level are declared Columbia 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 Columbia FSAR.

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. The NRC staff also notes that mechanical propeller style flow meters are not susceptible to radiation. The dose rates are provided in Table 2 of the FIP. The NRC staff confirmed the environmental information through audit reviews of Columbia document ME-02-14-04, “Reactor Building Accessibility and Equipment Operability during an ELAP.”

The NRC staff reviewed Section IV.C.10.3 and Table 2 of the FIP and determined the SAWA flow meter appears to be qualified for the anticipated environment. The NRC staff notes the location given in Table 1 of the FIP appears to be the storage location and not the point of use (reactor building El. 522) and did not rely on that information for this determination.

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 SAWA/SAWM Severe Accident Considerations

4.5.1 Staff Evaluation

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 FLEX building, which is protected from all screened in hazards. The SAWA pumps will be operated from outside the reactor building near the spray ponds. The licensee intends to establish SAWA prior to venting when there will be no issues with radiation dose rates at the SAWA pump control location and there will be no significant dose to the SAWA pump. After the HCV is opened SAWA pump will experience higher dose rates however anticipated dose rates should be well below that would impair SAWA pump operation.

In its FIP, the licensee stated that the SAWA flow consists of hoses that have been evaluated for the integrated dose effects over the period of sustained operation. 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 Columbia SAWA strategy relies on three instruments: wetwell level; containment pressure; and SAWA flow. Containment pressure and wetwell level are declared Columbia 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 expected to provide a continuous flow indication under the expected ambient conditions and therefor will be available for the entire period of sustained operation. Additionally, the licensee states in its FIP that the flow monitoring and adjustments are made in the reactor building stairwell at the 522-foot level. Anticipated dose rates at this location during HCV operation should be well below levels that would adversely impact the ability of the instruments to perform their intended function. 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

Columbia performed GOTHIC calculations to evaluate the expected temperature response of the reactor and control buildings during the ELAP event. Since, in the severe accident, the core materials are contained inside the primary containment, the temperature response of the control building is driven by the loss of ventilation and ambient conditions and therefore will not change as a result of the severe accident itself. In Section 3.1.1.2, “Personnel Habitability – Environmental,” above, the NRC staff provided its evaluation regarding potential environmental conditions for operating the hardened vent system. These conditions will not degrade during SAWA and SAWM.

In its FIP, the licensee stated that when implementing SAWA, personnel must deploy a portable DG. Actions performed in the reactor building will be completed within the first 6 hours. Environmental conditions for required actions performed outdoors, including access routes outside the reactor building, are not affected by a loss of ventilation during an ELAP.

Table 2 of Attachment 7 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 of the FIP shows the approximate locations of the actions.

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 ME-02-17-12, "Habitability and Access with Radiological Conditions Expected During an Extended Loss of AC Power with Severe Accident Conditions," which documents the dose assessment for designated areas inside the Columbia reactor building (outside of containment) and outside the Columbia 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 March 25, 2015 [Reference 16], an ISE for implementation of Phase 2 requirements on September 29, 2016 [Reference 17], and an audit report on the licensee's responses to the ISE open items on August 6, 2018 [Reference 18]. The licensee reached its final compliance date on June 28, 2019, and in letter dated August 20, 2019 [Reference 19], has declared that Columbia 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 Columbia to NRC, "Columbia Generating Station – 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. ML14191A688)

3. Letter from Columbia 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 17, 2014 (ADAMS Accession No. ML14357A069)

4. Letter from Columbia 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 30, 2015 (ADAMS Accession No. ML15181A436)

5. Letter from Columbia to NRC, "Columbia Generating Station – 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 16, 2015 (ADAMS Accession No. ML15351A363)

6. Letter from Columbia 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 30, 2016 (ADAMS Accession No. ML16182A080)

7. Letter from Columbia 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 29, 2016 (ADAMS Accession No. ML16364A245)

8. Letter from Columbia 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 27, 2017 (ADAMS Accession No. ML17178A276)

9. Letter from Columbia 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 28, 2017 (ADAMS Accession No. ML18002A438)

10. Letter from Columbia 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 21, 2018 (ADAMS Accession No. ML18176A186)

11. Letter from Columbia 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 13, 2018 (ADAMS Accession No. ML18347B495)

12. Letter from Columbia to NRC, "Final 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 24, 2019 (ADAMS Accession No. ML19175A271)

13. 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)

14. 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)

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

16. Letter from NRC to Columbia, "Columbia Generating Station – Interim Staff Evaluation Relating to Overall Integrated Plan in Response to Phase 1 of Order EA-13-109 (Severe Accident Capable Hardened Vents)," March 25, 2015 (ADAMS Accession No. ML14335A158)

17. Letter from NRC to Columbia, "Columbia Generating Station – 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 29, 2016 (ADAMS Accession No. ML16266A233)

18. Letter from NRC to Columbia, "Columbia Generating Station – 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 August 6, 2018 (ADAMS Accession No. ML18215A204)

19. Letter from Columbia to NRC, "Columbia Generating Station, "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 August 20, 2019 (ADAMS Accession No. ML19232A448)

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

21. 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)

22. 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)

23. Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," March 9, 2012 (ADAMS Accession No. ML12054A694)

24. SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments", November 26, 2012 (ADAMS Accession No. ML12325A704)

25. SRM-SECY-12-0157, "Staff Requirements - SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I And Mark II Containments", March 19, 2013 (ADAMS Accession No. ML13078A017)

26. NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 0, November 12, 2013 (ADAMS Accession No. ML13316A853)

27. Interim Staff Guidance JLD-ISG-2013-02, "Compliance with Order EA-13-109, Severe Accident Reliable Hardened Containment Vents," November 14, 2013 (ADAMS Accession No. ML13304B836)

28. NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 1, April 23, 2015 (ADAMS Accession No. ML15113B318)

29. Interim Staff Guidance JLD-ISG-2015-01, "Compliance with Phase 2 of Order EA-13-109, Severe Accident Reliable Hardened Containment Vents," April 29, 2015 (ADAMS Accession No. ML15104A118)

30. NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 2, December 2015 (ADAMS Accession No. ML16005A625)

31. NEI Industry White Paper – HCVS-WP-02, "Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping," Revision 0, October 30, 2014 (ADAMS Accession No. ML14358A038)

32. Letter from NEI to NRC, "Hardened Containment Venting System (HCVS) Phase 1 and 2 Overall Integrated Plan Template," Revision 1, dated September 28, 2015, and Frequently Asked Questions (FAQs) 10, 11, 12, and 13 (ADAMS Accession No. ML15273A141)

33. NEI Industry White Paper – HCVS-WP-04, "Tornado Missile Evaluation for HCVS Components 30 Feet Above Grade," Revision 0, August 17, 2015 (ADAMS Accession No. ML15244A923)

34. NEI Industry White Paper – HCVS-WP-03, "Hydrogen/Carbon Monoxide Control Measures," Revision 1, October 2014 (ADAMS Accession No. ML14295A442)

35. BWROG-TP-15-008, "BWROG Fukushima Response Committee, Severe Accident Water Addition Timing," September 2015

36. BWROG-TP-15-011, "BWROG Fukushima Response Committee, Severe Accident Water Management Supporting Evaluations," October 2015

37. NRC Letter to Columbia, "Columbia Generating Station – Safety Evaluation Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 And EA-12-051," February 22, 2018 (ADAMS Accession No. ML17333A888)

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