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

November 21, 2019

Mr. John A. Krakuszeski
Site Vice President
Brunswick Steam Electric Plant
8470 River Rd. SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NOS. MF4467 AND MF4468; EPID NO. L-2014-JLD-0041)

Dear Mr. Krakuszeski:

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 26, 2014 (ADAMS Accession No. ML14191A687), Duke Energy Progress LLC (Duke, the licensee), submitted its Phase 1 OIP for Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick) 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 Brunswick, including the combined Phase 1 and Phase 2 OIP in its letter dated December 11, 2015 (ADAMS Accession No. ML16020A064). 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 10, 2015 (Phase 1) (ADAMS Accession No. ML15049A266), August 17, 2016 (Phase 2) (ADAMS Accession No. ML16223A725), and March 22, 2018 (ADAMS Accession No. ML18068A627), the NRC issued Interim Staff Evaluations and an audit report, respectively, on the licensee's progress. By letter dated May 22, 2019 (ADAMS Accession No. ML19143A383), the licensee reported that Brunswick is in full compliance with the requirements of Order EA-13-109, and submitted a Final Integrated Plan for Brunswick.

The enclosed safety evaluation provides the results of the NRC staff's review of Brunswick’s hardened containment vent design and water management strategy for Brunswick. The intent of the safety evaluation is to inform Brunswick 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.

Sincerely,

/RA/

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

Docket Nos. 50-325 and 50-324

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ORDER EA-13-109

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

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 wet well 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 26, 2014 [Reference 2], Duke Energy Progress LLC (Duke, the licensee), submitted its Phase 1 Overall Integrated Plan (OIP) for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP, Brunswick) in response to Order EA-13-109. By letters dated December 17, 2014 [Reference 3], June 25, 2015 [Reference 4], December 11, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 28, 2016 [Reference 6], December 15, 2016 [Reference 7], June 19, 2017 [Reference 8], December 20, 2017 [Reference 9], June 26, 2018 [Reference 10], and December 19, 2018 [Reference 11], the licensee submitted 6-month updates to its OIP. By letters dated May 27, 2014 [Reference 12], and August 10, 2017 [Reference 13], the NRC notified all BWR Mark I and Mark II licensees that the staff will be conducting audits of their implementation of Order EA-13-109 in accordance with NRC Office of

Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, “Regulatory Audits” [Reference 14]. By letters dated March 10, 2015 (Phase 1) [Reference 15], August 17, 2016 (Phase 2) [Reference 16], and March 22, 2018 [Reference 17], the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated May 22, 2019 [Reference 18], the licensee reported that full compliance with the requirements of Order EA-13-109 was achieved and submitted its Final Integrated Plan (FIP).

2.0 REGULATORY EVALUATION

Following the events at the Fukushima Dai-ichi nuclear power plant on March 11, 2011, the NRC established a senior-level agency task force referred to as the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of the NRC regulations and processes and determining if the agency should make improvements to these programs in light of the events at Fukushima Dai-ichi. As a result of this review, the NTTF developed a set of recommendations, documented in SECY-11-0093, “Near-Term Report and Recommendations for Agency Actions Following the Events in Japan,” dated July 12, 2011 [Reference 19]. Following interactions with stakeholders, these recommendations were enhanced by the NRC staff and presented to the Commission.

On February 17, 2012 [Reference 20], the NRC staff provided SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan’s March 11, 2011, Great Tohoku Earthquake and Tsunami,” to the Commission. This paper included a proposal to order licensees to implement the installation of a reliable hardened containment venting system (HCVS) for Mark I and Mark II containments. As directed by the Commission in staff requirements memorandum (SRM)-SECY-12-0025 [Reference 21], the NRC staff issued Order EA-12-050, “Order Modifying Licenses with Regard to Reliable Hardened Containment Vents” [Reference 22], which required licensees to install a reliable HCVS for Mark I and Mark II containments.

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

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

2.1 Order EA-13-109, Phase 1

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

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

2.2 Order EA-13-109, Phase 2

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

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

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

Brunswick is a two-unit site and each unit is a General Electric BWR with a Mark I primary containment system. Containment integrity is maintained by controlling containment pressure using the HCVS. The HCVS is initiated using manual action from the 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 29], that are applicable to the plant site. Operator actions to initiate the HCVS vent path can be completed by plant personnel and include the capability for remote-manual initiation from the HCVS control station. A list of the remote manual actions performed by plant personnel to open the HCVS vent path are listed in Table 3-1, "HCVS Operator Actions," of the FIP. An HCVS extended loss of alternating current (ac) power (ELAP) Failure Evaluation Table (FIP Table 3-2), which shows alternate actions that can be performed, is also provided in the FIP.

The licensee also stated that permanently-installed electrical power and pneumatic supplies are available to support operation and monitoring of the HCVS for a minimum of 24 hours. No large portable equipment needs to be moved in the first 24 hours to operate the HCVS. After 24 hours, available personnel will be able to connect supplemental electric power and pneumatic supplies for sustained operation of the HCVS for a minimum of 7 days. The FLEX generators and air compressors provide this motive force. Likely, these actions will be completed in less than 24 hours. However, the HCVS can be operated for at least 24 hours without any supplementation.

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 (SOV) failure.

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

3.1.1.2 Personnel Habitability – Environmental

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

In its FIP, the licensee stated that primary control of the HCVS is accomplished from the main control room (MCR). Alternate control of the HCVS is accomplished from the remote operating station (ROS) at the 50' elevation (second floor) of the reactor building. FLEX actions that will maintain the MCR and ROS habitable were implemented in response to NRC Order EA-12-049. These actions include:

1. Restoring MCR ventilation via the FLEX diesel generator. MCR ventilation was included as a load in the FLEX generator sizing calculations and is acceptable.
2. Opening selected MCR doors (if required).
3. Operating portable generators and fans to move outside air through the MCR (if required).
4. Opening doors and a roof hatch in the reactor building to establish natural circulation air flow at the ROS in the reactor building.

Table 2 of the FIP contains a thermal evaluation of all the operator actions that may be required to support HCVS operation. Calculations BNP-MECH-FLEX-0001, "Reactor Building FLEX Heat-up," and BNP-MECH-FLEX-0003, "Control Building FLEX Room Heat-up," demonstrates that the final design meets the order requirements to minimize the plant operators' exposure to occupational hazards.

The NRC staff audited calculations RWA-L-1312-003, BNP-MECH-FLEX-0001 and BNP-MECH-FLEX-0003. Calculation RWA-L-1312-003 uses the Generation of Containment Thermal-Hydraulic Information (GOTHIC) computer program to model the control building heat-up during the ELAP event. The control room temperature peaks at 116 degrees Fahrenheit (°F). After compensatory actions are taken to open selected doors and install a portable fan, the temperature drops below 95°F for the remainder of the 72 hours modeled. Calculation BNP-MECH-FLEX-0001 also uses the GOTHIC computer program to model the reactor building during the ELAP event. The ROS (50' elevation in the reactor building) peaks at 121°F for the 7-day period with the mitigating actions of opening doors and the roof hatch. The NRC staff also noted that the stay times in the ROS are limited and strenuous work tasks are not required to be performed to accomplish needed tasks and agrees that the temperatures in the MCR and ROS should not inhibit operators from performing their required tasks.

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 EC 412141, "Integrated Dose Calculation for Hardened Vent Remote Operating Station (ROS) – Fukushima," documents the dose assessment for designated areas inside the BSEP reactor building (outside of containment) and outside the BSEP reactor building caused by the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. Calculation EC 412141 was performed using NRC-endorsed HCVS-WP-02 [Reference 30] and HCVS-FAQ-12 [Reference 31] methodologies. Consistent with the definition of sustained operations in NEI 13-02, Revision 1, the integrated whole-body gamma dose equivalent[1] due to HCVS operation over a 7-day period as determined in the licensee's dose calculation should not exceed 10 Roentgen equivalent man (rem)[2]. The calculated 7-day dose due to HCVS operation is a conservative maximum integrated radiation dose over a 7-day period with ELAP and fuel failure starting at reactor shutdown. For the sources considered and the methodology used in the calculation, the timing of HCVS vent operation or cycling of the vent will not create higher doses at personnel habitability and equipment locations (i.e., maximum doses determined in the calculation bound operational considerations for HCVS vent operation).

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

The licensee evaluated the maximum dose rates and 7-day integrated whole-body gamma dose equivalents for the MCR, which is the primary control location and the ROS. In its FIP, the licensee states that the ROS location and the travel path to the 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 EC 412141) demonstrates that the

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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

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 on the 50' elevation of the reactor building, just inside the door to the radwaste building roof. The licensee stated that the ROS location is in an area evaluated to be accessible before and during a severe accident. The licensee also provided, in Table 1 of its FIP, a list of the controls and indications that are or may be required to operate the HCVS during a severe accident, including the locations, anticipated environmental conditions, and the environmental conditions (temperature and radiation) to which each component is qualified.

The NRC staff reviewed the FIP including the response in Section III.B.1.1.4 of the FIP and examined the information provided in Table 1. The NRC staff determined that the controls and indications appear to be consistent with the NEI 13-02 guidance. The NRC staff also confirmed the environmental qualification information in Table 1 of the FIP through audit reviews of Brunswick calculation BNP-MECH-FLEX-0001, "RB FLEX Room Heat-Up," Revision 0. The NRC staff noted that the Regulatory Guide (RG) 1.97 instruments for drywell pressure and wetwell level did not include some qualification information in Table 1, but are considered acceptable, in accordance with the NEI 13-02 guidance, based on the original qualification for severe accident conditions.

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

3.1.2 Design Features

Order EA-13-109 requires that the HCVS shall include specific design features, including specifications of the vent characteristics, vent path and discharge, 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 0FLEX-0035, "Flow Capacity of BNP Hardened Wetwell Vent Units 1 & 2 at 1% Rated Power," Revision 0, which provides the verification of the capability to vent the steam/energy equivalent of one percent rated reactor thermal power at design pressure (62 pounds per square inch gauge (psig)), which is lower that the primary containment pressure limit (PCPL) (70 psig). This calculation models all the piping elbows, valves and other components using industry standard flow coefficients to determine an equivalent length of piping. Since the piping consists of 8" and 12" sections, both are modeled. The model is input to the ARROW code, which is an industry standard program for modeling compressible flow in piping. The code also looks for flow choking effects. The minimum flow at design pressure to pass one percent reactor thermal power is 84,500 pounds mass per hour (lbm/hr). Calculation 0FLEX-0035 verifies that the piping can pass greater than one percent flow. Additional assumptions and modeling details are contained in calculation 0FLEX-0035.

The decay heat absorbing capacity of the suppression pool and the selection of venting pressure were made such that the HCVS will have sufficient capacity to maintain containment pressure at the containment design pressure (62 psig). This calculation of containment response is contained in Modular Accident Analysis Program (MAAP) calculation BNP-MECH-FLEX-0002, “Brunswick Nuclear Plant Containment Analysis of FLEX Strategies,” Revision 0, which shows that containment is maintained below the design pressure once the vent is opened, even if it is not opened until PCPL.

The NRC staff reviewed the information provided and audited calculation 0FLEX-0035. The calculation assumed a rated reactor thermal power of 2,923 megawatts thermal (MWt). The calculation shows the required vent capacity is 84,420 lbm/hr. The HCVS vent pipe system resistance was determined using a flow of 85,000 lbm/hr. Based on the evaluation, the HCVS vent design appears to have the capacity to vent one percent of rated thermal power during ELAP and severe accident conditions with margin.

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

3.1.2.2 Vent Path and Discharge

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

The NRC staff evaluated the HCVS vent path and the location of the discharge. The vent exits the primary containment through the wetwell purge exhaust piping and associated inboard wetwell purge exhaust valve. Between the inboard and outboard wetwell purge exhaust valves, the wetwell vent isolation valve is installed. Downstream of the wetwell vent isolation valve, the vent piping exits the reactor building through the west wall and into the space between the reactor building and turbine building. The vent traverses up the exterior of the building and re-enters the reactor building through the metal siding on the refuel floor, then rises along the west side where it exits the reactor building through the roof. All effluents are exhausted above each unit’s reactor building. This discharge point was extended approximately 3 feet above each unit’s reactor building parapet wall. The NRC staff’s review indicates that this appears to be consistent with the guidance provided in HCVS-FAQ-04.

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

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

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

1. For the portions of exposed piping below 30 feet above grade, the pipe is located in a narrow access space between two concrete walls where it is highly unlikely that a missile could penetrate to a depth required to damage the vent pipe.

2. The exposed piping greater than 30 feet above grade has the following characteristics:
   a. The total vent pipe exposed area is 200 square feet which is less than the 300 square feet discussed in HCVS-WP-04.
   b. The pipe is made of schedule 40 stainless steel and the pipe components have no small tubing susceptible to missiles.
   c. There are no obvious sources of missiles located in the proximity of the exposed HCVS components.

3. BSEP maintains severe weather preparedness procedures that would require the plant be shut down prior to the arrival of sustained hurricane force winds on site.

The NRC staff audited calculation PGB024-CALC-002, "Evaluation of the Hardened Wetwell Vent for Beyond Design-Basis Events," Revision 0. The calculation indicated that the hardened wetwell vent (HWV) is routed through a seismic isolation space between the reactor building and the turbine building. The HWV is protected from tornado missiles by the reactor building and the turbine building. The HWV re-enters the reactor building and exits through the roof. The reactor building provides tornado missile protection. The licensee looked at potential wind-driven missiles and concluded that none of the potential missiles could be accelerated to speeds significant to damage the HWV. The NRC staff agrees that the HCVS pipe is adequately protected against seismic or tornado events.

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

3.1.2.3 Unintended Cross Flow of Vented Fluids

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

In its FIP, the licensee stated that the HCVS for Units 1 and 2 are fully independent of each other. Therefore, the status of each unit is independent of the status of the other unit.

The wetwell vent for each unit utilizes containment atmosphere control system (CAC) valves CAC-V7 and CAC-V216 for containment isolation. The CAC system containment isolation valves CAC-V8 and CAC-V172 are the only functional boundary valves between the HCVS and the downstream standby gas treatment system (SBGT). These valves have a safety-related function to maintain the containment pressure boundary during a design basis accident and are tested as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The containment isolation valves are air-operated valves (AOVs). They are air-to-open and spring-to-shut. An SOV must be energized to allow the motive air to open the valve from the MCR location. Although these valves are shared between the CAC and the HCVS, separate control circuits are provided to each valve. Specifically, the CAC control circuit will be used during all "design basis" operating modes including all design basis transients and accidents. The downstream CAC isolation valves serve the function of isolating the HCVS flow path from the SBGT. These valves are tested, and will continue to be tested, for leakage under 10 CFR Part 50, Appendix J as part of the containment boundary in accordance with HCVS-FAQ-05. The NRC staff audited the information provided and agrees that the use of primary containment isolation valves appears to be acceptable for prevention of inadvertent cross-flow of vented fluids and consistent with the guidance provided in HCVS-FAQ-05.

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

3.1.2.4 Control Panels

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

In its FIP, the licensee stated that the existing wetwell vent will allow initiating and then operating and monitoring from 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 to meet the requirement for an alternate means of operation, a readily accessible alternate location, called the ROS was added for each unit and is located on the 50' elevation (second floor) of the reactor building. Each ROS contains manually operated valves that supply pneumatics to the HCVS flow path valve actuators and rupture disk so that the HCVS may be opened without power to the valve actuator solenoids and regardless of any containment isolation signals that may be actuated. This provides a diverse method of valve operation and improves system reliability. Attachment 2 of the FIP shows the valves and flow path used for manual operation.

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

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

3.1.2.6 Power and Pneumatic Supply Sources

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

<u>Pneumatic Sources Analysis</u>

For the first 24 hours following the ELAP event, the nitrogen backup system will automatically provide operating pneumatics to the safety relief valve (SRV) accumulators and HWV valves. The existing nitrogen backup system bottle racks are located in the reactor building. Two additional nitrogen bottles have been added to the nitrogen backup system to provide motive force for up to 20 venting cycles, which includes opening the vent valves (CAC-V7 and CAC-V216) for 24 hours. In its FIP, the licensee stated that 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 0RNA-0001, "Instrument Air Nitrogen Backup System Volume Requirements," Revision 4. The licensee's calculation determined that the expanded 12 bottle nitrogen backup system (originally 10 bottles) will provide sufficient motive force to open CAC-V7 once and will open CAC-V216 20 times. The licensee's calculation determined that a minimum cylinder pressure of 1,130 psig, as specified by Brunswick Technical Specification SR 3.6.1.5.1, can provide sufficient capacity for operation of the HCVS valves for 24 hours following an ELAP and the SRVs for the required number of times per the design basis. This pressure includes an allowance for leakage.

The NRC staff audited the calculation and confirmed that there should be sufficient pneumatic supply available to provide motive force to operate the HCVS 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, Brunswick would rely on the existing Units 1 and Unit 2, Division 2, 24/48 volt (V) direct current (dc) battery to supply HCVS loads. The batteries are permanently installed in the control building on the ground floor where they are protected from applicable hazards.

The Division 2, 24/48 Vdc battery is model Gould NCX-9 with a nominal capacity of 558 ampere hours (Ah). The Division 2, 24/48 Vdc battery has a minimum capacity capable of providing power for 24 hours without recharging. During the audit period, the licensee provided the NRC staff with an evaluation for the Division 2, 24/48 Vdc battery requirements including incorporation into the FLEX diesel generator (DG) loading calculation.

The NRC staff audited licensee calculations BNP-E-6.128 (Unit 1), “24/48 VDC Battery Allowable Discharge Rate for HCVS during an ELAP,” Revision 0 and BNP-E-6.125 (Unit 2), “24/48 VDC Battery Allowable Discharge Rate for HCVS during an ELAP,” Revision 0, which verified the capability of the Division 2, 24/48 Vdc batteries to supply power to the required loads during the first phase of the Brunswick venting strategy for an ELAP. Battery sizing worksheet from IEEE Standard 485-2010, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” which is endorsed by RG 1.212, “Sizing of Large Lead-Acid Storage Batteries,” published in 2015 was used to determine if there were adequate positive plates available to power the existing plant loads and HCVS loads for 24 hours. The licensee’s calculation identified the required loads and their associated ratings (combined current and minimum system operating voltage). The licensee’s battery sizing calculation showed that a maximum 7 amperes of continuous loading for a 24-hour duty period. The Division 2, 24/48 Vdc battery has 149 percent of design margin available. Therefore, the Brunswick Division 2, 24/48 Vdc batteries should have sufficient capacity to supply power for at least 24 hours.

The licensee’s strategy includes repowering 480 Vac buses within 1 hour after initiation of an ELAP event using permanently pre-staged 500-kilowatt (kW) (one per unit) 480 Vac FLEX DGs. The 480 Vac FLEX DGs would provide power to the HCVS load in addition of loads addressed under Order EA-12-049.

The NRC also staff audited licensee calculation 3116-CALC-E-001, “FLEX Diesel Generator Sizing Calculation,” Revision 0, under Order EA-12-049. The calculation shows that 150 kW is required for the Division II loads from each unit, for a total required capacity of 300 kW. Based on the margin available for the 500 kW 480 Vac FLEX DGs, the NRC staff finds that one FLEX DG has sufficient capacity and capability to supply the required electrical loads including HCVS loads at both units.

Electrical Connection Points

The licensee’s strategy to supply power to HCVS components requires using a combination of permanently installed and portable components. Connecting the permanently pre-staged 500 kW FLEX DG were addressed under Order EA-12-049.

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

The containment isolation valves must be open to permit vent flow. The physical features that prevent inadvertent actuation are the key lock switch for CAC-V216 at the primary control station and locked closed valves at the ROS. These design features meet the requirement to prevent inadvertent actuation of HCVS. The NRC staff's audit confirmed that the licensee's design is consistent with the guidance and appears to preclude inadvertent actuation.

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

3.1.2.8 Monitoring of HCVS

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

The NRC staff reviewed the following channels documented in Table 1 of the FIP that support HCVS operation: HCVS effluent temperature; HCVS battery voltage; wetwell vent line radiation; HCVS valve position; N_2 pressure (mechanical); drywell pressure; and wetwell level. The NRC staff notes that wetwell level is a declared Brunswick post-accident monitoring (PAM) variable as described in RG 1.97 and the existing qualification of this channel 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 notes that the drywell pressure instrument channel is not a designated RG 1.97 variable, but the licensee stated that the channel is similarly qualified, and the NRC staff reviewed the drywell pressure qualification information provided in Table 1 of the FIP and found it appears to meet the guidance. The NRC staff also reviewed FIP Section

III.B.1.2.8 and determined that the HCVS instrumentation appears to be adequate to support HCVS venting operations and is capable of performing its intended function during ELAP and severe accident conditions.

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

3.1.2.9 Monitoring of Effluent Discharge

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

The NRC staff reviewed the following channels documented in Table 1 of the FIP which support monitoring of HCVS effluent: HCVS valve position (PCIV limit switch contacts); HCVS effluent temperature; and wetwell 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 and Table 1 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 that is fully qualified for the expected environment at the vent pipe during accident conditions, and the process and control module is qualified for the environment in the MCR. The NRC staff audited the qualification summary information provided in Table 1 of the FIP and found that it appeared to meet the guidance.

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

3.1.2.10 Equipment Operability (Environmental/Radiological)

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

<u>Environmental</u>

The FLEX diesel driven SAWA pumps are commercial fire pumps rated for long-term outdoor use in emergency scenarios. The SAWA pumps will be staged outside so they will not be adversely impacted by a loss of ventilation. The FLEX DGs are permanently pre-staged in vented enclosures and are not impacted by loss of ventilation.

As discussed above in Section 3.1.1.2, the licensee performed calculation BNP-MECH-FLEX-0003, which predicts the temperature profile of the MCR following an ELAP. The licensee determined that performing compensatory ventilation actions (opening doors and establishing portable ventilation) will maintain temperatures in the MCR area below 95°F. Licensee procedure OEOP-01-SB0-02, "Blacked Out Unit Initial Actions," Revision 1, directs operators to open doors and establish portable ventilation.

The Division 2, 24/48 Vdc batteries are installed in the control building on the 23' elevation. Licensee calculation BNP-MECH-FLEX-0003 predicts the temperature profile on the 23' elevation in the control building following an ELAP. The licensee conservatively determined that performing compensatory ventilation actions (opening doors) on the 23' elevation of the control building will maintain temperatures in the battery room area below 110°F. The licensee can also restore normal battery room ventilation when the permanently pre-staged FLEX DGs are repowering the 480 Vac buses which will further reduce temperature in this area. Licensee procedure 0EOP-01-FSG-04, "FLEX Diesel Generator Alignment," Revision 3, directs operators to open doors and restore battery room ventilation.

Based on the above, the NRC staff concurs with the licensee's calculations that show the control building 23' elevation in the vicinity of the battery rooms will remain below the maximum temperature limit (120°F) of the Division 2, 24/48 Vdc batteries, as specified by the battery manufacturer (Exide Technologies). Furthermore, based on temperature remaining below 120°F (the temperature limit for electronic equipment to be able to survive indefinitely, identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, as endorsed by NRC RG 1.155), the NRC staff believes that electrical equipment located in the MCR should not be adversely impacted by the loss of ventilation as a result of an ELAP event. 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 EC 412141, "Integrated Dose Calculation for Hardened Vent Remote Operating Station (ROS) – Fukushima," 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 EC 412141 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 3.2.1.2, the CAC system containment isolation valves CAC-V8 and CAC-V172 are the only functional boundary valves between the HCVS and the downstream SBGT. These valves have a safety-related function to maintain the containment pressure boundary during a design basis accident and are tested as required by 10 CFR Part 50, Appendix J. The NRC staff's audit confirmed that the design appears to be consistent with the guidance and meets the design requirements to minimize the potential of hydrogen gas migration into other buildings.

Based on the evaluation above, the NRC staff concludes that the licensee's HCVS design minimizes the potential for hydrogen gas migration and ingress, and, if implemented appropriately, appears to be consistent with NEI 13-02 guidance, as endorsed by

JLD-ISG-2013-02 and JLD-ISG-2015-01, and should adequately address the requirements of the order.

3.1.2.13 HCVS Operation/Testing/Inspection/Maintenance

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

In the Brunswick 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 outboard containment isolation valve (CAV-V216) and penetrations were not modified for order compliance so that they continue to be designed consistent with the design basis of primary containment including pressure, temperature, radiation, and seismic loads. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators, and containment isolation valve position indication components. The hardened vent piping, between the wetwell and the reactor building roof, including boundary isolation valve (CAC-V8) are designed to 70 psig at 316°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 Brunswick PCPL is 70 psig with a corresponding saturation temperature of 316°F. Brunswick used a design value of 316°F for the vent piping, corresponding to the saturation temperature for the BSEP PCPL of 70 psig. Thus, the temperature of 316°F will be retained as the pipe design temperature. This lower HCVS design temperature is adequate for component qualifications, since it is acceptable (per the guidance in NEI 13-02 Rev. 1, Section 2.4.3.1) to assume saturation conditions in containment.

The HCVS 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 its 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 instrumentation components required for HCVS venting are designed to remain functional following a design basis earthquake.

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

3.2.2 Component Reliability and Rugged Performance

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

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

As part of the NRC staff's audit, the NRC staff requested information verifying that existing containment isolation valves, relied upon for HCVS operation, will open under the maximum expected differential pressure during a BDBEE and severe accident wetwell venting. The licensee provided procedure 0EOP-02-PCCP, "Primary Containment Control Procedure," which directs opening the hardened wetwell vent valves before reaching the PCPL of 70 psig. Therefore, the maximum opening differential pressure is 70 pounds per square inch differential (psid) (containment to atmosphere). Calculation BNP-MECH-AOV-DP-CAC, "Differential Pressure Calculations for ½ -CAC-V7-AO, -V8-AO, -V216-AO – Inboard Suppression Pool Purge Exhaust, Outboard Suppression Pool Purge Exhaust, Outboard Suppression Pool Purge Exhaust, and Hardened Wetwell Vent Isolation Air-Operated Valves," Revision 0, discusses the valve/actuator information for the PCIVs. The calculation demonstrates that valves will open under a maximum differential pressure of 70 psid.

Calculations BNP-MECH-1-CAC-V7-AO, "AOV Setup Calculation for 1-CAC-V7-AO Torus Purge Exhaust Valve," Revision 1 and BNP-MECH-2-CAC-V7-AO, "AOV Setup Calculation for

2-CAC-V7-AO Torus Purge Exhaust Valve," Revision 1, contain the AOV calculations for the inboard wetwell purge valve on each unit. Section 4.1.1 of the calculation contains a table of minimum margins for these valves. The minimum opening margin for 1-CAC-V7 is 12.7% and for 2-CAC-V7 is 19.5%

Calculations BNP-MECH-1-CAC-V216-AO, "AOV Setup Calculation for 1-CAC-V216-AO Hardened Wetwell Vent Outboard Isolation Valve," Revision 1 and BNP-MECH-2-CAC-V216-AO, "AOV Setup Calculation for 2-CAC-V216-AO Hardened Wetwell Vent Outboard Isolation Valve," Revision 1, contain the AOV calculations for the hardened wetwell vent valve on each unit. Section 4.1.1 of the calculation contains a table of minimum margins for these valves. The minimum opening margin for 1-CAC-216 is 33.8 percent and for 2-CAC-216 is 25.7 percent.

The NRC staff's audit verified the actuator can develop greater torque than the PCIVs unseating torque. Therefore, the PCIVs should open under the maximum expected differential pressure during beyond-design-basis and severe accident wetwell venting.

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

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

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

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

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

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

- The strategy making it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions shall be part of the overall accident management plan for Mark I and Mark II containments;

- The licensee shall provide supporting documentation demonstrating that containment failure as a result of overpressure can be prevented without a drywell vent during severe accident conditions; and,

- Implementation of the strategy shall include licensees preparing the necessary procedures, defining and fulfilling functional requirements for installed or portable equipment (e.g., pumps and valves), and installing the needed instrumentation.

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

4.1 Severe Accident Water Addition (SAWA)

The licensee plans to use the portable diesel-driven FLEX pump to provide SAWA flow. The SAWA flow path is the same as the FLEX primary injection flow path. The water source will be either the condensate storage tank (CST) or the discharge canal. The SAWA flow path hose routing is from the FLEX pump to the RPV. Most of the SAWA actions take place outside the reactor building and 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, using one of the manifolds described below.

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 were replaced with stainless steel pipe so operators did not have to route house in the reactor building during severe accident conditions. The SAWA flow path's primary water source is the CST with a backup suction source from the discharge canal. The primary SAWA flow path hose routing is from the FLEX pump to a core bore in the reactor building. From the core bore, stainless steel pipe connects to the reactor water cleanup (RWCU) system, which then ties into the reactor core isolation cooling (RCIC) and then to the RPV via the reactor feedwater system. Backflow prevention is provided by existing, safety-related, check valves installed in the RWCU and feedwater system, 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 on the pumps. The SAWA flow indication is placed near the pump in the discharge piping.

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 strategies. The licensee described the hydraulic analysis performed to demonstrate the capability of FLEX pumps to provide the required 300 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 0FLEX-0003, "Hydraulic Analysis for Fukushima FLEX Connection Modifications," Revision 3, which determined that each of these three pumps would be able to provide the required SAWA flow rate of 300 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 Brunswick, 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 flow.

4.1.1.3 SAWA Analysis of Flow Rates and Timing

The licensee developed the overall accident management plan for Brunswick 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 BSEP severe accident management guidelines (SAMGs). The EPG/SAG Revision 3, when implemented with emergency procedures committee Generic Issue 1314, allows throttling of SAWA valves to protect containment while maintaining the wetwell vent in service. The SAMG flow charts direct use of the hardened vent as well as SAWA/SAWM when the appropriate plant conditions have been reached.

The licensee used NEI 12-06, Appendix E to validate the FLEX water system pumps used for SAWA can be deployed and commence injection in less than 8 hours. The studies referenced in NEI 13-02 demonstrated that establishing flow within 8 hours will protect containment. Guidance document NEI 13-02, Appendix I, establishes an initial water addition rate of 500 gpm based on EPRI Technical Report 3002003301, “Technical Basis for Severe-Accident Mitigating Strategies.” The initial SAWA flow rate at Brunswick will be at least 300 gpm based on the site's rated thermal power compared to the reference power level in NEI 13-02. After roughly 4 hours, during which the maximum flow rate is maintained, the SAWA flow will be reduced. The reduction in flow rate and the timing of the reduction will be based on stabilization of the containment parameters of drywell pressure and torus level.

Calculation BNP-MECH-FLEX-005, “Brunswick Nuclear Plant MAAP 5.02 Analysis to Support SAWA Strategy,” Revision 1 demonstrates that SAWA flow could be reduced to 100 gpm after 4 hours of initial SAWA flow rate and containment would remain protected. At some point, if wetwell level begins to rise, indicating that the SAWA flow is greater than the steaming rate due to containment heat load, SAWA flow can be further reduced as directed by the SAMGs.

In its FIP, the licensee stated that the torus vent was designed and installed to meet NEI 13-02, Revision 1, guidance and is sized to prevent containment overpressure under severe accident conditions. The licensee will follow the guidance (flow rate and timing) for SAWA described in BWROG-TP-15-008, “Severe Accident Water Addition Timing,” [Reference 35] and BWROG-TP-15-011, “Severe Accident Water Management” [Reference 36]. The wetwell vent will be opened prior to exceeding the PCPL value of 70 psig. The licensee also referenced analysis included in BWROG-TP-15-008, which demonstrates adding water to the reactor vessel within 8 hours of the onset of the event will limit the peak containment drywell temperature, significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the containment from the suppression chamber.

The NRC staff audited calculation BNP-MECH-FLEX-0005. Guidance document NEI 13-02, uses an initial SAWA flow of 500 gpm reduced to 100 gpm after 4 hours. Brunswick evaluated a 300 gpm SAWA flow starting at 8 hours. At 12 hours SAWA flow is reduced to 100 gpm. The NRC staff noted that the licensee determined plant-specific flow rates using the ratio of Brunswick licensed thermal power (2923 MWt) to that of the reference plant (3,514 MWt) in the EPRI Technical Report 3002003301, "Technical Basis for Severe-Accident Mitigating Strategies." This is consistent with NEI 13-02, Section 4.1.1.2.

4.1.2 Conclusions

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

4.2 Severe Accident Water Management

The licensee's strategy to preclude the necessity for installing a hardened drywell vent at Brunswick is to implement the containment venting strategy utilizing SAWA and SAWM. This strategy consists of the use of the Phase 1 torus vent and SAWA hardware to implement a water management strategy that will preserve the torus vent path until alternate reliable containment heat removal can be established. The SAWA system consists of a FLEX pump injecting into the RPV. The overall strategy consists of flow control by throttling valves at the FLEX (SAWA) pump along with instrumentation and procedures to ensure that the wetwell vent is not submerged (SAWM). Water from the SAWA (FLEX) pump will be routed to a core bore in the reactor building which connects to the RWCU system. This RWCU connection ties into the reactor core isolation cooling (RCIC) system and then to the RPV via the reactor feedwater system. 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 MAAP analysis to protect containment without requiring a drywell vent for at least seven days, which is the guidance from NEI 13-02 for the period of sustained operation.

4.2.1 Staff Evaluation

4.2.1.1 Available Freeboard Use

In its FIP, the licensee states that the freeboard between elevations -2.5' and +6' in the wetwell provides approximately 536,700 gallons of water volume before the level instrument would be off scale high. 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. Brunswick performed calculation BNP-MECH-FLEX-0005, demonstrating that the wetwell level will not reach the wetwell vent for at least 7 days.

As noted above in Section 4.1.1.3, the NRC staff audited calculation BNP-MECH-FLEX-0005, Revision 1. Brunswick evaluated a 300 gpm SAWA flow starting at 8 hours. At 12 hours SAWA flow is reduced to 100 gpm. The calculation assumed a license thermal power of 2923 MWt. The calculation concludes that drywell temperature peaks at roughly 581°F and drops below 300° after SAWA flow is initiated. With 300 gpm SAWA followed by 100 gpm SAWA flow starting at 12 hours and continuing for 168 hours will not result in a large increase in the suppression pool water lever that could potentially challenge the operation of the HWV. The NRC staff concurs that the flow of water added to the suppression pool can be controlled such that the suppression pool remains operational.

4.2.1.2 Strategy Time Line

As noted in Section 4.1.1.3, "SAWA Analysis of Flow Rates and Timing," the SAWA flow is based on calculation BNP-MECH-FLEX-0005 and BWROG-TP-15-011 to demonstrate that throttling SAWA flow after containment parameters have stabilized, in conjunction with venting containment through the torus vent will result in a stable or slowly rising torus level. The references demonstrate that, for the scenario analyzed, wetwell level will remain below the upper range of the wetwell level instrument, and below the wetwell vent pipe for greater than the 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 8hours 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 three portable pumps to provide SAWA flow to each unit (1 pump per unit). The pumps are diesel-driven by an engine mounted on the skid with the pump. Operators will refuel the pumps' DGs in accordance with Order EA-12-049 procedures using fuel oil from the installed, underground DG fuel oil storage tanks. In its FIP, the licensee states that refueling will be accomplished in areas that are shielded and protected from the radiological conditions during a severe accident scenario. 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 lists drywell pressure, wetwell level, and the SAWA flow meter, as instruments required for SAWA and SAWM implementation. The wetwell level and drywell pressure are used for HCVS venting operation. These instruments are powered by the Division

2, 24/48 Vdc batteries until the FLEX DGs are available to repower battery chargers. The SAWA flow meter is a paddle-wheel flow meter powered by the pump's electrical system and provides indication of flow rate for the operator at the pump.

The NRC staff audited licensee calculations BNP-E-6.128 (Unit 1) and BNP-E-6.125 (Unit 2), which verified the capability of the Division 2, 24/48 Vdc batteries to supply power to the required loads (e.g., wetwell level and drywell pressure) during the first phase of the Brunswick FLEX mitigation strategy plan for an ELAP event. The NRC staff also audited licensee calculation 3116-CALC-E-001, which verified that the 500 kW FLEX DG is adequate to support the other SAWA/SAWM electrical loads. The NRC staff confirmed that Division 2, 24/48 Vdc batteries and 500 kW FLEX DGs should have sufficient capacity and capability to supply the necessary SAWA/SAWM loads during an ELAP event.

4.3.2 Conclusions

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

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

4.4.1 <u>Staff Evaluation</u>

4.4.1.1 SAWA/SAWM Instruments

In Section IV.C.10.2 of its FIP, the licensee stated that the instrumentation needed to implement the SAWA/SAWM strategy are wetwell level, drywell pressure, and SAWA flow. The NRC staff found that wetwell level is an existing RG 1.97 instrument that was designed and qualified for severe accident conditions. Drywell pressure is not a RG 1.97 instrument, but is qualified for the anticipated environment as indicated in Table 1 of the FIP. The licensee also stated, in Table 1 of its FIP, that SAWA flow instrument range is 0 to 500 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, 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 Brunswick strategy may also make use of drywell temperature. The licensee also stated that SAMG strategies will evaluate and use drywell temperature indication if available consistent with the symptom-based approach.

In Section IV.C.10.2 of its FIP, the licensee stated that the SAWA flow meter is a paddle-wheel mounted in piping on the pump's skid and provides indication of flow rate for the operator at the pump.

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

4.4.1.3 Qualification of SAWA/SAWM Instruments

In Section IV.C.10.3 of its FIP, the licensee stated that wetwell level is a declared Brunswick PAM variable as described in RG 1.97 and the existing qualification of this channel 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 variable in the Brunswick Final Safety Analysis Report. The NRC staff notes that the licensee clarified, in the Table 1 of the FIP, that drywell pressure is not an RG 1.97 qualified variable, but is qualified for the anticipated environment, as reviewed in Section 3.1.2.8, above.

In its FIP, the licensee stated that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. Furthermore, since the pump is deployed outside on the opposite side of the reactor building from the vent pipe, it is well shielded, by structure and distance, from high radiation. The NRC staff also notes that mechanical paddle-wheel style flow meters are not susceptible to radiation. The NRC staff reviewed Section IV.C.10.3 of the FIP and determined the SAWA flow meter appears to be qualified for the anticipated environment.

4.4.2 Conclusions

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

4.5 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 FSB and will be operated from outside the reactor building, on the opposite side of the reactor building from the vent pipe. Therefore, there will be no significant issues with radiation dose rates at the SAWA pump control location and there will be no significant dose to the SAWA pump.

In its FIP, the licensee stated that the SAWA flow path inside the reactor building consists of steel piping that will be unaffected by the anticipated radiation dose rates. The NRC staff audited the information and agrees that the SAWA flow path will not be adversely affected by radiation effects due to the severe accident conditions.

4.5.1.2 Severe Accident Effect on SAWA/SAWM Instruments

The BSEP SAWA strategy relies on three instruments: wetwell level; drywell pressure; and SAWA flow. Wetwell level is a declared BSEP PAM variable as described in RG 1.97 and the existing qualification of this channel 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. Containment pressure is not an RG 1.97 qualified variable, but is qualified for the anticipated environment, as reviewed in Section 3.1.2.8, above.

In its FIP, the licensee states that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. Additionally, the licensee states in its FIP that since the SAWA pump is deployed outside the reactor building on the opposite of the reactor building from the vent pipe, the effects of radiation exposure on the flow instrument should be minimal. 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

The Brunswick FIP notes that actions inside the reactor building are accomplished within the first 7 hours. These actions, including access to routes outside the reactor building that will be performed after the first use of the vent during severe accident conditions (assumed to be 7 hours per HCVS-FAQ-12), are located such that they are either shielded from direct exposure to the vent line or are a significant distance from the vent line, so that expected dose is maintained below the ERO exposure guidelines.

As part of the response to Order EA-12-049, BSEP performed GOTHIC calculations of the temperature response of the reactor and control buildings during the ELAP event. In a severe accident, the core materials are contained inside the primary containment, therefore the temperature response of the reactor building and control building, which is driven by the loss of ventilation and ambient conditions are not expected to change.

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

After the SAWA pipe is aligned inside the reactor building, the operators can control SAWA/SAWM as well as observe the necessary instruments from outside the reactor building. The thick concrete reactor building walls (below 117' level), as well as the distance to the core materials, mean that there is no radiological concern with any actions outside the reactor building. Therefore, all SAWA controls and indications are accessible during severe accident conditions.

The SAWA monitoring is from the MCR or from outside the reactor building at ground level. The SAWA pump operation is from outside the reactor building at ground level. The BSEP FLEX response ensures that the SAWA pump, FLEX air compressors, FLEX generators and other equipment can all be run for a sustained period by refueling. All the refueling locations are in shielded or protected areas so that there is no radiation hazard from core material during a severe accident. The monitoring instrumentation includes SAWA flow at the pump, and wetwell level and containment pressure in the MCR.

The NRC staff noted that environmental conditions reviewed in Section 3.1.1.2, “Personnel Habitability – Environmental” do not change during SAWA/SAWM operation. The NRC staff also reviewed the actions and environmental conditions documented in Table 2 of the FIP. Existing plant guidance should provide protection for operators performing work outdoors. The NRC staff concludes that the projected environmental conditions for monitoring and operating the SAWA/SAWM strategy will not prevent operators from performing required actions to implement that plan.

The licensee performed calculation EC 412141, “Integrated Dose Calculation for Hardened Vent Remote Operating Station (ROS) – Fukushima," which documents the dose assessment for designated areas inside the BSEP reactor building (outside of containment) and outside the BSEP reactor building caused by FLEX activities and the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. This assessment used conservative assumptions to determine the expected dose rates in all areas that may require access during a beyond-design-basis ELAP. As stated in Section 3.1.1.3, “Personnel Habitability - Radiological,” the NRC staff agrees, based on the audit of the licensee's detailed evaluation, that mission doses associated with actions taken to protect the public under beyond-design-basis severe accident conditions will not subject plant personnel to an undue risk from radiation exposure.

4.5.2 Conclusions

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

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

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

5.0 HCVS/SAWA/SAWM PROGRAMMATIC CONTROLS

5.1 Procedures

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

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

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

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

5.2 Training

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

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

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

6.0 CONCLUSION

In June 2014, the NRC staff started audits of the licensee's progress in complying with Order EA-13-109. The staff issued an ISE for implementation of Phase 1 requirements on April 2, 2015 [Reference 15], an ISE for implementation of Phase 2 requirements on September 6, 2016 [Reference 16], and an audit report on the licensee's responses to the ISE open items on April 10, 2018 [Reference 17]. The licensee reached its final compliance date on April 25, 2019, and in letter dated April 25, 2019 [Reference 17], has declared that Brunswick 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 Brunswick to NRC, “Brunswick Steam Electric Plant, Units 1 and 2 – 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 26, 2014 (ADAMS Accession No. ML14191A687)

3. Letter from Brunswick 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. ML14364A029)

4. Letter from Brunswick 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 25, 2015 (ADAMS Accession No. ML15196A035)

5. Letter from Brunswick to NRC, “Brunswick Steam Electric Plant, Units 1 and 2 – 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 11, 2015 (ADAMS Accession No. ML16020A064)

6. Letter from Brunswick 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 28, 2016 (ADAMS Accession No. ML16190A111)

7. Letter from Brunswick 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 15, 2016 (ADAMS Accession No. ML16365A007)

8. Letter from Brunswick 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 19, 2017 (ADAMS Accession No. ML17171A383)

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Date: November 21, 2019

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NOS. MF4467 AND MF4468; EPID NO. L-2014-JLD-0041) DATED NOVEMBER 21, 2019

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