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

January 2, 2020

Mr. Joseph W. Shea
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NOS. MF4540, MF4541, AND MF4542; EPID NO. L-2014-JLD-0044)

Dear Mr. Shea:

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

By letter dated June 30, 2014 (ADAMS Accession No. ML14181B169), Tennessee Valley Authority (TVA, the licensee), submitted its Phase 1 OIP for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry) 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 Browns Ferry, including the combined Phase 1 and Phase 2 OIP in its letter dated December 29, 2015 (ADAMS Accession No. ML15365A554). 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 February 11, 2015 (Phase 1) (ADAMS Accession No. ML14356A362), September 6, 2016 (Phase 2) (ADAMS Accession No. ML16244A762), and February 21, 2018 (ADAMS Accession No. ML18038B606), the NRC issued Interim Staff Evaluations and an audit report, respectively, on the licensee's progress. By letter dated June 7, 2019 (ADAMS Accession No. ML19157A297), the licensee reported that Browns Ferry is in full compliance with the requirements of Order EA-13-109, and submitted a Final Integrated Plan for Browns Ferry.

The enclosed safety evaluation provides the results of the NRC staff's review of Browns Ferry's hardened containment vent design and water management strategy for Browns Ferry. The intent of the safety evaluation is to inform Browns Ferry 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. 50-259, 50-260, and 50-296

Enclosure:
Safety Evaluation

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NOS. MF4540, MF4541, AND MF4542; EPID NO. L-2014-JLD-0044)

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ORDER EA-13-109

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

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

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

By letter dated June 30, 2014 [Reference 2], Tennessee Valley Authority (TVA, the licensee), submitted its Phase 1 Overall Integrated Plan (OIP) for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry) in response to Order EA-13-109. By letters dated December 19, 2014 [Reference 3], June 29, 2015 [Reference 4], December 29, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 30, 2016 [Reference 6], December 22, 2016 [Reference 7], June 30, 2017 [Reference 8], December 20, 2017 [Reference 9], June 27, 2018 [Reference 10], and December 27, 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 February 11, 2015 (Phase 1) [Reference 15], September 6, 2016 (Phase 2) [Reference 16], and February 21, 2018 [Reference 17], the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated June 7, 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

Browns Ferry is a three-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 diesel generator (DG) and nitrogen bottles 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) located in the diesel generator buildings. 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 480-Volt FLEX diesel generator sizing calculations and is acceptable.

2. Opening selected doors in various buildings to establish natural circulation at the ROS.

3. Operating portable diesel generators and fans to move outside air through the MCR (if required).

Table 2 of the FIP contains a thermal evaluation of all the operator actions that may be required to support HCVS operation. Calculations MDQ0009992014000291, "Temperature Response of the Reactor Building Following an Extended Loss of AC Power" and MDQ0003602014000222, "BFN ELAP Transient Temperature Analysis," demonstrate that the final design meets the order requirements to minimize the plant operators' exposure to occupational hazards.

The NRC staff audited the calculations and design change packages. The calculations use the Generation of Thermal-Hydraulic Information in Containment (GOTHIC) computer program to model building heat-up during the ELAP event. Compensatory actions of opening select doors at 1-hour and several other doors opening at 12-hours into the event are modeled in the calculations.

Licensee calculation MDQ0003602014000222, "BFN ELAP Transient Temperature Analysis" shows that with compensatory actions the MCRs for Unit-1, Unit-2, and Unit-3 remain below 110 degrees Fahrenheit (°F) at 72 hours. Calculation MDQ0030880213, "Unit 1 and Unit 2 DGB - Central Diesel Information Center Ventilation Requirements," determines normal ventilation requirements for the ROS area (Units 1 and 2) during normal operation. During ELAP, normal ventilation is not available. However, prior to an ELAP, there is no normal operating equipment which will provide a residual heat load. Given the mass of concrete construction around the ROS along with no major electrical heat loads or residual heat loads from operating equipment, area temperatures will not be averse to operators performing their required actions in the ROS.

Calculation MDN0009992012000027, “Thermal Analysis of Control Bay Rooms, Unit 3 DGB SDBRs and BR4BR Following Loss of Cooling,” evaluated the Unit-3 ROS area in the Unit-3 diesel generator building and the shutdown board room (SDBR). The calculation predicts the temperature in these areas remains below 110°F.

During the audit, the NRC staff noted a fire protection Cardox tank located near the ROS. During an ELAP, cooling will be lost to the Cardox tank which could lead to a pressure increase in the tank. The licensee confirmed the CO_2 is vented outside the building where it cannot re-enter the building and will not impact the ROS area.

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 MDQ0000642015000351, “HCVS Operator (Mission) Dose Calculation,” documents the dose assessment for designated areas inside the Browns Ferry reactor building (outside of containment) and outside the Browns Ferry reactor building caused by the sustained operation of the HCVS under the beyond-design-basis severe accident condition of an ELAP. Calculation MDQ0000642015000351 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

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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

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 MDQ0000642015000351) demonstrates that the integrated whole-body gamma dose equivalent to personnel occupying defined habitability locations (resulting from HCVS operation under beyond-design-basis severe accident conditions) should not exceed 10 rem.

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

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

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

3.1.1.4 HCVS Controls and Indications

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

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

In Section III.B.1.1.4 of its FIP, the licensee stated that primary control of the HCVS is accomplished from the MCR and that under the postulated scenarios of Order EA-13-109, the

MCR is adequately protected from excessive radiation dose and no further evaluation of its use is required (HCVS-FAQ-06). Alternate control of the HCVS is accomplished from the ROS in the Unit 1/2 and Unit 3 DGBs. The licensee stated the ROS locations are in areas evaluated to be accessible before and during a severe accident. The licensee also provided, in Table 1 of the FIP, a list of the controls and indications that are or may be required to operate the HCVS during a severe accident, including the locations, anticipated environmental conditions, and the environmental conditions (temperature and radiation) to which each component is qualified.

The NRC staff reviewed the FIP including the response in Section III.B.1.1.4 of the FIP and examined the information provided in Table 1. The NRC staff determined that the controls and indications appear to be consistent with the NEI 13-02 guidance. The NRC staff also confirmed the environmental qualification information in Table 1 of the FIP through audit reviews of Browns Ferry calculation MDQ0000642015000351, "HCVS Operator Mission Dose," and calculation MDQ0009992014000291, "Temperature Response of the Reactor Building Following and ELAP." 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 documented its Modular Accident Analysis Program (MAAP) analysis in calculation NDQ0000642015000341 for the final design of HCVS, which covers all three units at Browns Ferry. The calculation provides verification of one percent power flow capacity at design pressure (62 pounds per square inch gauge (psig)) and also validates that Browns Ferry's suppression pools can absorb the decay heat generated for the first 3 hours following a shutdown using decay heat values associated with the uprated thermal power value of 3952 mega-watt thermal (MWt). In addition, MAAP generates mass flow rates based on a range of assumed discharge coefficients to show that the as designed HCVS can relieve the effects of decay heat during the transient. Browns Ferry calculations NDQ0010642015000342 for Unit 1, NDQ0020642015000400 for Unit 2, and NDQ0030642015000399 for Unit 3 perform analyses to determine flow rates and the actual discharge coefficients for the HCVS piping configuration based on set wetwell pressures. The calculations use the GOTHIC computer code. Once the actual discharge coefficient was determined, it was verified to be within the range specified by MAAP in Browns Ferry Calculation NDQ0000642015000341. The MAAP analysis concludes that the chosen vent size is adequate to maintain primary containment temperatures and pressures below 350°F and 62 psig while venting steam/energy equivalent to one percent of the uprated thermal power. In making this conclusion, MAAP considers both the successful implementation of FLEX, as well as early and latent failures of the reactor core isolation cooling (RCIC) system which results in a severe accident. Severe accident scenarios considered by MAAP include scenarios where core debris is cooled in the reactor vessel and external to the reactor vessel.

The decay heat absorbing capacity of the suppression pool and the selection of venting pressure were made such that the HCVS will have the capacity to maintain containment pressure at or below the lower of the containment design pressure (56 psig) or the primary containment pressure limit (PCPL) (62 psig). This calculation of containment response is contained in HCVS MAAP Analysis NDQ0000642015000341.

The NRC staff reviewed the information provided and audited calculation NDQ0000642015000341. The calculation assumed a rated reactor thermal power of 3,952 MWt. At 56 psig, the corresponding required rate of flow for 1 percent rated thermal power is 31.7 pounds (mass) per second (lbm/sec). The calculation determined the HCVS flow rate at this pressure is 58.4 lbm/sec, which is greater than the minimum required flow. 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 wetwell vent exits the primary containment by using portions of the original hardened wetwell vent (HWWV) piping inside the Units 1, 2, and 3 reactor buildings and associated inboard and outboard primary containment isolation valves (PCIVs). New HCVS piping travels vertically up the exterior wall of the Units 1, 2, and 3 reactor buildings. All new HCVS piping is 14 inches schedule 40 while the existing HWWV piping is 14 inches schedule 30. The piping continues up the exterior wall of the Units 1, 2, and 3 reactor buildings until it reaches an approximate elevation of 665 feet where it turns to penetrate the siding of the Units 1, 2, and 3 reactor building superstructures. Once inside the superstructure the piping continues vertically until it penetrates the roof of the superstructure, ultimately terminating at approximate elevation 741 feet and 6 inches. The vent pipe extends approximately 3 feet 6 inches above the existing vent tower elevation of 738 feet, the highest existing elevation of the reactor building roof.

This is consistent with the guidance provided in HCVS-FAQ-04. The vent system is designed to preclude Hydrogen/Carbon Monoxide detonation by placement of a vent discharge check valve at the approximate elevation of the Units 1, 2, and 3 reactor building roof. The potential for lightning strike is mitigated through the installation of a lightning protection device installed on the HCVS piping near the vent release point.

Part of the HCVS-FAQ-04 clarifies the guidance to ensure that vented fluids are not drawn immediately back into any emergency ventilation intakes. Such ventilation intakes should be below a level of the pipe by 1 foot for every 5 horizontal feet. The chosen release point is situated away from the MCR ventilation system intake and exhaust openings. Therefore, the vent pipe is appropriately placed relative to this air intake.

Guidance document NEI 13-02, Section 5.1.1.6, provides guidance that missile impacts are to be considered for portions of the HCVS. The NRC-endorsed NEI white paper, HCVS-WP-04, "Missile Evaluation for HCVS Components 30 Feet Above Grade," Revision 0 [Reference 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 supplemental guidance 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 various Browns Ferry site areas were reviewed for their potential to create missiles, defined by NRC RG 1.76 Revision 1, dated March 2007, which may strike unprotected HCVS piping and components located less than 30 feet above grade. The review was performed to validate the first assumption from NEI White Paper HCVS-WP-04. It has been determined that it is not credible that any tornado borne commodities within the scope of the first assumption will strike and jeopardize function of the HCVS. This review and conclusions are documented in Browns Ferry White Paper “Validation of NEI White Paper HCVS-WP-04 First Assumption for Missile Protection of Hardened Containment Vent System at BFN.”

2. The exposed piping greater than 30 feet above grade has the following characteristics:
    a. The total vent pipe exposed area is not in excess of 250 square feet (ft^2) which results in a potential missile target area less than the 300 ft^2 limit specified in HCVS-WP-04.
    b. The pipe is made of schedule 40 carbon steel and is not plastic 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. Browns Ferry maintains a large cutoff saw as part of the FLEX equipment. This saw can cut the vent pipe should it become damaged such that it restricts flow to an unacceptable level.

4. Hurricanes are not screened for Browns Ferry.

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, 2, and 3 are fully independent of each other. Therefore, the status at each unit is independent of the status of the other unit. The cross flow within the unit is prevented by boundary valves that are considered primary containment isolation valves. These valves are:

- 1,2,3-FCV-064-0019 Inboard Valve Cooling/Purge Air to Torus
- 1,2,3-FCV-064-0020 Inboard Valve Torus Vacuum Relief
- 1,2,3-FCV-064-0021 Inboard Valve Torus Vacuum Relief
- 1,2,3-FCV-064-0221 Inboard Valve HCVS
- 1,2,3-FCV-064-0222 Outboard Valve HCVS
- 1,2,3-FSV-84-0008B Inboard Valve CAD Admission to Torus
- 1,2,3-FSV-084-0008C Inboard Valve CAD Admission to Torus
- 1,2,3-FSV-076-0019 Inboard Valve Containment Inerting, N_2 Makeup

(Where CAD stands for Containment Atmosphere Dilution system and N_2 stands for nitrogen)

The NRC staff reviewed the information provided and noted that the boundary valves are described in the final safety analysis report (FSAR) Table 5.2-2, "Principle Primary Containment Penetrations and Associated Isolation Valves." The primary containment isolation valves are subject to leak testing under the guidance of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," [Reference 33] test program. 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, 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.

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, the ROS, was added. The location for the ROS is in the Unit 1/2 and Unit 3 diesel generator buildings, which similar to the main control room, is also protected from environmental conditions related to operation of HCVS. The ROS contains manually operated valves that supply pneumatic power directly to the HCVS flow path valve actuators so that these valves 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 therefore improving system reliability.

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

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

3.1.2.6 Power and Pneumatic Supply Sources

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

<u>Pneumatic Sources Analysis</u>

For the first 24 hours following the ELAP event, the pneumatic motive force for the actuation of the two PCIVs in each Unit will be supplied by two nitrogen air bottles per Unit. Units 2 and 3, if operated simultaneously, will require 5 nitrogen bottles over a 24-hour period. The nitrogen air bottles are connected to the ROS within the Unit 1, 2, and 3 diesel generator buildings. These bottles have been sized such that they can provide motive force for eight vent cycles of the PCIVs.

The licensee determined the required pneumatic supply storage volume and supply pressure set point required to operate the PCIVs actuation for 24 hours following a loss of normal pneumatic supplies during an ELAP in calculation MDQ0000322015000347, "HCVS Nitrogen Sizing Analysis," Revision 1. The minimum required pressure for total HCVS operation is calculated at around 100 psig for each unit. The licensee's calculation determined that two nitrogen bottles per unit, each filled at the maximum capacity of 2400 psig, will provide sufficient capacity for eight cycles of the PCIVs for 24 hours following an ELAP. This pressure includes an allowance for leakage. The NRC staff audited the calculations and confirmed that there should be sufficient pneumatic supply available to provide motive force to operate the HCVS system for 24 hours following a loss of normal pneumatic supplies during an ELAP.

<u>Power Source Analysis</u>

In its FIP, the licensee stated that during the first 24 hours of an ELAP event, Browns Ferry would rely on a new dedicated battery and battery charger with sufficient capacity to supply HCVS loads. The Unit 1 250-volt (V) direct current (dc) HCVS battery, battery charger, distribution panel, and manual transfer switches are in the Unit 1/2 diesel building. The Unit 2 and Unit 3 250 Vdc HCVS battery, battery charger, distribution panel, and manual transfer switches are common to both units and are in the Unit 3 diesel building. The HCVS batteries and battery chargers are installed where they are protected from applicable hazards. Exide Technologies manufactured the HCVS battery.

The HCVS batteries are model GNB Absolyte GP 6-50G05 with a nominal capacity of 122.4 ampere hours (Ah). The HCVS battery has a minimum capacity capable of providing power for

24 hours without recharging. During the audit period, the licensee provided the NRC staff an evaluation for the HCVS battery/battery charger sizing requirements. The licensee does not plan to recharge the HCVS batteries after depletion, therefore the HCVS battery chargers were not incorporated into the FLEX combustion turbine generator (CTG) loading calculation. The licensee plans to transfer the HCVS electrical loads back to their normal power supply, which is the Class 1E station batteries.

The NRC staff audited licensee calculations EDQ0010642015000349, “Unit 1 HCVS Electrical Design & Equipment Sizing Analysis,” Revision 2 and EDQ0000642016000510, “Unit 2 & 3 HCVS DC Electrical Design & Equipment Sizing Analysis,” Revision 0, which verified the capability of the HCVS battery to supply power to the required loads during the first phase of the Browns Ferry venting strategy for an ELAP. The HCVS battery was sized in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 485-2010, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” which is endorsed by RG 1.212, “Sizing of Large Lead-Acid Storage Batteries,” published 2015. The licensee’s calculation identified the required loads and their associated ratings (current, watts, and minimum operating voltage). The minimum battery requirement for Unit 1 HCVS loads is 37.4 Ah, with a current capability of at least 1.56 A at 24 hours. The minimum battery requirement for Unit 2 and Unit 3 HCVS loads is 85.22 Ah, with a current capability of at least 3.55 A at 24 hours. The HCVS battery chosen has a 24-hour rating of 5.1 A and 122.4 Ah. Therefore, the Browns Ferry HCVS battery should have sufficient capacity to supply power for at least 24 hours.

The licensee’s strategy includes transitioning power to the Class 1E 250 Vdc station batteries within 24 hours, which would be energized by its battery chargers powered by a FLEX CTG. The licensee’s strategy relies on 4160 Volt alternating current (Vac) or 480 Vac (alternate strategy) FLEX CTGs to provide power to the HCVS load in addition of loads addressed under Order EA-12-049.

The NRC staff audited licensee calculation EDQ0003602015000325, "Electrical Evaluation for 4kV FLEX Turbine Generators,” Revision 1. The calculation shows that the loading for the 1 megawatt (MW) 4160 Vac FLEX CTG is 862 kilo-volt ampere (kVA) which includes the Class 1E 250 Vdc station battery chargers. The rating of the 4KV FLEX CTG is 1250 kVA. The NRC staff also audited licensee calculation EDQ0003602014000281, "Electrical Evaluation for Portable Power Supply for Unit Battery Chargers,” Revision 1. This calculation shows the loading for the 825 kilowatt (kW) 480 Vac FLEX CTG is 489 kW which also include the Class 1E 250 Vdc station battery chargers. Therefore, sufficient margin exists on the 1 MW and 825 kW FLEX CTGs to power HCVS loads.

Electrical Connection Points

The licensee’s strategy to supply power to HCVS components requires using a combination of permanently installed and portable components. Staging and connecting the 1 MW 4160 Vac and 825 kW 480 Vac FLEX CTGs 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 provide clear guidance that the HCVS is not to be used to defeat containment integrity during any design basis transients and accidents. In addition, the HCVS was designed to provide features to prevent inadvertent actuation due to equipment malfunction or operator error.

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

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

3.1.2.8 Monitoring of HCVS

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

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

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

3.1.2.9 Monitoring of Effluent Discharge

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

The NRC staff reviewed the following channels documented in Table 1 of the FIP, which supports monitoring of HCVS effluent: HCVS valve position; 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 Units 1, 2, and 3 computer rooms on elevation 1C of the control building. The NRC staff audited the qualification summary information provided in Table 1 of the FIP and found that it appeared to meet the guidance.

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

3.1.2.10 Equipment Operability (Environmental/Radiological)

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

Environmental

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

As discussed above in Section 3.1.1.2, the alternate control of the HCVS is accomplished from the ROS. All actions at the ROS are manual and do not require any electrical power.

The HCVS battery, battery charger, and supporting equipment for Unit 1 are permanently installed in the Units 1 and 2 diesel generator building in the electrical access room. The HCVS battery, battery charger, and supporting equipment for Units 2 and 3 are permanently installed in the Unit 3 diesel generator building in the shutdown board room. As discussed above in Section 3.1.1.2, the licensee performed calculations MDQ0030880213 (Units 1 and 2) and MDN0009992012000027 (Unit 3) which predicts the temperature profile in the respective diesel generator building. The licensee determined that performing compensatory actions (opening doors and establishing ventilation) in the areas will maintain temperature less than 110°F. Based on the above, the NRC staff concurs with the licensee's calculations that show that the Units 1, 2, and 3 diesel generator building will remain below the maximum temperature limit (120°F) of HCVS batteries, as specified by the battery manufacturer (Exide Technologies). Furthermore, based on temperature remaining below 120°F (the temperature limit for electronic equipment to be able to survive indefinitely, identified in NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, as endorsed by NRC RG 1.155), the NRC staff believes that other electrical equipment located in the area should not be adversely impacted by the loss of ventilation as a result of an ELAP event with the HCVS in operation. Therefore, the NRC staff concurs that the HCVS equipment should not be adversely impacted by the loss of ventilation as a result of an ELAP event.

Radiological

The licensee's calculation MDQ0000642015000351, "HCVS Operator (Mission) Dose Calculation," 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 MDQ0000642015000351 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.

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

3.1.2.11 Hydrogen Combustible Control

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

Guidance document NEI 13-02, Section 4.1.7 provides guidance for the protection from flammable gas deflagration/detonation in the HCVS. The NEI issued white paper HCVS-WP-03, "Hydrogen/Carbon Monoxide Control Measures," Revision 1, endorsed by the NRC [Reference 34], which provides methods to address control of flammable gases.

One of the acceptable methods described in the white paper is the installation of a check valve at or near the end of the vent stack to restrict the ingress of air to the vent pipe when venting stops and steam condenses (Option 5).

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

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

3.1.2.12 Hydrogen Migration and Ingress

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

As discussed in Section 3.1.2.3, "Unintended Cross Flow of Vented Fluids," the units are independent of each other. Connection to other systems in the same unit is precluded by primary containment isolation valves which are routinely 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 Browns Ferry FIP, Table 3-3 includes testing and inspection requirements for HCVS components. The NRC staff reviewed Table 3-3 and confirmed that it is consistent with Section 6.2.4 of NEI 13-02, Revision 1. Implementation of these testing and inspection requirements for the HCVS will ensure reliable operation of the systems.

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

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

3.2 HCVS QUALITY STANDARDS

3.2.1 Component Qualifications

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

In its FIP, the licensee stated that the HCVS upstream of and including the second containment isolation valve (1,2,3-FCV-064-0222) and penetrations are not being modified for order compliance so that they continue to be designed consistent with the design basis of primary containment including pressure, temperature, radiation, and seismic loads. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators, and containment isolation valve position indication components. The hardened vent piping between the wetwell and the reactor building roof is designed to 62 psig at 350°F. Guidance document NEI 13-02 suggests a 350°F value for HCVS design temperature based on the highest PCPL among the Mark I and II plants.

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

The licensee further provided Table 1 in 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 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 the 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 structures and that all instruments are fully qualified for the expected seismic conditions so that they will remain functional following a seismic event.

In its FIP, the licensee indicated that boundary valves design temperature and pressures were evaluated against the HCVS design pressure (62 psig) and temperature (350°F) to determine if the boundary valve can remain closed during HCVS operation.

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 performed an evaluation and concluded that the containment isolation valves will open under the maximum expected differential pressure and is documented in FLOWSERVE Report RAL-70181, "Design Review Report of Size 14, Class 150, Wafer Butterfly Valve with Pneumatic Actuator" Revision 1. Evaluation RAL-70181 determine that the maximum allowable stem torque is 6000 in-lbf and the maximum pressure differential across the containment isolation valves is 70.7 psi. The operating torque at the valve seat is expected to increase approximately 11 percent due to the increase in differential pressure from 56 psi to 70.7 psi. The calculated required torque to start open increases from 4944 inch-pounds (in-lbs.) to 5508 in-lbs. The actuator is a Bettis model NCB725-SR80-MCW pneumatic quarter turn actuator with an internal coil spring to fail close and air pressure to open. The actuator start-to-open output torque varies from 5064 in-lbs. at 70 pounds psig actuator air pressure to 6395 in-lbs. at 80 psig air pressure. An air pressure of 75 psig will provide a start to open torque of approximately 5730 in-lbs. sufficient to open the valve at the higher differential pressure.

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. Browns Ferry has elected the option to develop and implement a reliable containment venting strategy that makes it unlikely the licensee would need to vent from the containment drywell before alternate reliable containment heat removal and pressure control is reestablished.

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

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

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

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

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

4.1 Severe Accident Water Addition (SAWA)

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

4.1.1 Staff Evaluation

4.1.1.1 Flow Path

The SAWA injection flow path starts with the FLEX Triton pump suctioning from the Tennessee River and going into the FLEX Dominator pump to a flow meter trailer and a flexible discharge hose. The SAWA flowpath continues from the flow meter trailer to the containment integrated leak rate test (CILRT) connection, which penetrates the reactor building wall. The piping from the reactor building wall is connected to the condensate storage supply (CS&S) system piping, which supplies core spray loops I and II, which is then injected to the RPV using the core spray injection valves. The hoses and FLEX pumps used for SAWA flow are stored in the FLEX equipment storage building (FESB) building, which is protected from all external hazards. This SAWA injection path is also protected from all applicable external hazards in addition to severe accident conditions.

4.1.1.2 SAWA Pump

In its FIP, the licensee states that the strategy is to use one of four sets of the FLEX Triton and Dominator pumps, connected in series, for FLEX and SAWA strategies. The FLEX pumps are trailer-mounted and are both portable diesel-driven pumps. The FLEX pumps will inject SAWA flow into the RPV within 8 hours of the ELAP event and provides 500 gallons per minute (gpm) of SAWA flow for the first 4 hours of operation. The FLEX pumps can supply each unit with 500 gpm, with the fire hose connections attached from the discharge portions of the FLEX Dominator pump. The SAWA flow will be reduced to 100 gpm for the duration of the ELAP event. All three sets of the FLEX pumps are stored in the FESB, which is protected from all applicable external hazards. In its FIP, the licensee described the hydraulic analysis performed to demonstrate the capability of the FLEX pumps connected in series to provide the required SAWA flow. The NRC staff audited calculation MDN0003602014000233, "Hydraulic Analysis for Fukushima FLEX Connection Modifications," Revision 3, which determined that the required SAWA flow rate of 500 gpm per unit was within the capacity of the FLEX pumps.

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

4.1.1.3 SAWA Analysis of Flow Rates and Timing

The licensee developed the overall accident management plan for Browns Ferry 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 Browns Ferry severe accident management guidelines (SAMGs). The EPG/SAG Revision 3, when implemented with emergency procedures committee Generic Issue 1314, allows throttling of SAWA in order 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 that the SAWA pump 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 Browns Ferry will be at least 500 gpm. After a period, estimated to be about 4 hours, in which the maximum flow rate is maintained, the SAWA flow will be reduced. The reduction in flow rate and the timing of the reduction will be based on stabilization of the containment parameters of drywell pressure and wetwell level. Guidance document NEI 13-02 generic analysis demonstrated that, SAWA flow could be reduced to 100 gpm after 4 hours of initial SAWA flow rate and containment would be protected. At some point, if wetwell level begins to rise, indicating that the SAWA flow is greater than the steaming rate due to containment heat load, SAWA flow can be further reduced as directed by the SAMGs.

In its FIP, the licensee stated that the torus vent was designed and installed to meet NEI 13-02, Revision 1, guidance and is sized to prevent containment overpressure under severe accident conditions. The licensee will follow the guidance (flow rate and timing) for SAWA described in BWROG-TP-15-008, "Severe Accident Water Addition Timing," [Reference 35] and BWROG-

TP-15-011, "Severe Accident Water Management" [Reference 36]. The wetwell vent will be opened prior to exceeding the PCPL value of 62 psig. The licensee also referenced analysis included in BWROG-TP-15-008, which demonstrates adding water to the reactor vessel within 8 hours of the onset of the event will limit the peak containment drywell temperature, significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the containment from the suppression chamber.

Browns Ferry SAWA flow is 500 gpm, which is the amount assumed in the guidance of NEI 13-02, Section 4.1.1.2.1. The initial SAWA flow will be injecting to the RPV within 8 hours of the loss of injection. The reference power level is 3514 MWt, equivalent to the reference plant rated thermal power level used in NUREG-1935, "State of the Art Reactor Consequence Analysis (SOARCA)." The SAWA and SAWM industry study (The EPRI study "Technical Basis for Severe Accident Mitigating Strategies, 3002003301" assumes a 500 gpm SAWA injection flow) was based on a reference plant which has the most limiting containment heat capacity in the U.S. fleet and therefore is conservative.

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 Browns Ferry 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 pump along with instrumentation and procedures to ensure that the wetwell vent is not submerged. Water from the FLEX pump will be routed to the CILRT connection which penetrates the reactor building wall. The CILRT connection ties into the CS&S system which supplies core spray loops I and II and then to the RPV via the core spray injection valves. 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 and Emergency Procedures Committee Generic Issue 1314. This strategy has been shown via MAAP analysis to protect containment without requiring a drywell vent for at least 7 days, which is the guidance from NEI 13-02 for the period of sustained operation.

4.2.1 Staff Evaluation

4.2.1.1 Available Freeboard Use

In its FIP, the licensee states that the freeboard between elevations -1 inch and 26.3 feet in the wetwell provides approximately 757,544 gallons of water volume before the water level reaches the bottom of the vent pipe. A diagram of the available freeboard is shown on Attachment 1 to the FIP.

Generic assessment BWROG-TP-15-011, provides the principles of SAWM to preserve the wetwell vent for a minimum of 7 days. After containment parameters are stabilized with SAWA flow, SAWA flow will be reduced to a point where containment pressure will remain low while wetwell level is stable or very slowly rising. Browns Ferry performed analysis 32-9248484-003, "Browns Ferry Nuclear Power Plant (BFN) severe Accident Water Addition (SAWA) Analysis," demonstrating that the wetwell level will not reach the wetwell vent for at least 7 days.

The NRC staff audited the information provided and agrees that starting the water addition at the higher flow rate and throttling back after approximately 4-hours will not increase the suppression pool level to a point that could block the suppression chamber HCVS opening before operators can take additional actions to maintain containment integrity.

4.2.1.2 Strategy Time Line

As noted above, the SAWA flow is based on calculation 32-9248484-003 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 8-hours of the onset of the event will limit the peak containment drywell temperature significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the suppression chamber through the suppression pool. Technical Paper BWROG-TP-011 demonstrates that starting water addition at a high rate of flow and throttling after approximately 4 hours will not increase the suppression pool level to that which could block the suppression chamber HCVS.

4.2.2 <u>Conclusions</u>

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

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

4.3.1 <u>Staff Evaluation</u>

4.3.1.1 SAWA Pump Power Source

As described in Section 4.1, the licensee plans to use the FLEX Triton and Dominator pumps to provide SAWA flow. Operators will refuel the FLEX pumps in accordance with Order EA-12-049 procedures using diesel fuel from the installed emergency diesel generator (EDG) fuel oil storage tanks. FLEX support instruction 0-FSI-6B, “FLEX Long Term Fueling Operations," Revision 0, directs operators to refuel the portable FLEX pumps from the onsite plant diesel generator 7-day tanks or onsite 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.

4.3.1.2 DG Loading Calculation for SAWA/SAWM Equipment

In its FIP, the licensee list drywell pressure, wetwell level, and the portable SAWA flow meter, as instruments required for SAWA and SAWM implementation. The wetwell level and drywell pressure are used for HCVS venting operation. These instruments are powered by the Class 1E 250 Vdc station batteries until the FLEX CTG is deployed and available. The SAWA flow meter is a paddle-wheel flow meter powered by the diesel-driven Dominator SAWA pump electrical system.

The NRC staff audited licensee dc coping calculation EDQ0009992013000202, “250V DC Unit Batteries, 1, 2, & 3 Evaluation for the Beyond Design Basis External Event (BDBEE) Extended Loss of AC Power (ELAP),” Revision 1, under Order EA-12-049, which verified the capability of the Class 1E 250 Vdc station batteries to supply power to the required loads (e.g., wetwell level and drywell pressure) during the first phase of the Browns Ferry FLEX mitigation strategy plan for an ELAP event. The NRC staff also audited licensee calculations EDQ0003602015000325 and EDQ0003602014000281, which verified that the 1 MW and 825 kW FLEX CTGs, respectively, are adequate to support the SAWA/SAWM electrical loads. The NRC staff confirmed that the Class 1E 250 Vdc station batteries, and 1 MW and 825 kW FLEX CTGs should have sufficient capacity and capability to supply the necessary SAWA/SAWM loads during an ELAP event.

4.3.2 Conclusions

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

4.4 SAWA/SAWM Instrumentation

4.4.1 Staff Evaluation

4.4.1.1 SAWA/SAWM Instruments

In Section IV.C.10.2 of its FIP, the licensee stated that the instrumentation needed to implement the SAWA/SAWM strategy are wetwell level, drywell pressure, and the SAWA flow meter. The NRC staff found that drywell pressure and wetwell level are existing RG 1.97 instruments that were designed and qualified for severe accident conditions. The licensee also stated, in Table 1 of its FIP, that SAWA flow instrument range is 40 to 1380 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 Browns Ferry 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 flow indicator trailer and is powered by the pump's electrical system.

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 the drywell pressure and wetwell level are declared Browns Ferry PAM variables as described in RG 1.97 and the existing qualification of these channels is considered acceptable for compliance with Order EA-13-109 in accordance with the guidance in NEI 13-02, Appendix C, Section C.8.1. The NRC staff verified the RG 1.97 variables in the Browns Ferry FSAR.

In its FIP, the licensee stated that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. Furthermore, since the pump is deployed outside of the reactor building and a significant distance from the vent pipe, there is no concern for any effects of radiation on the flow instrument. 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 <u>SAWA/SAWM Severe Accident Considerations</u>

4.5.1 <u>Staff Evaluation</u>

4.5.1.1 Severe Accident Effect on SAWA Pump and Flowpath

In its FIP, the licensee stated that the FLEX pumps are stored in the FESB and are protected from all screened in hazards. The pumps will be operated from outside the reactor building, at a significant distance from the vent pipe. Therefore, there will be no issues with radiation dose rates at the SAWA pump control location and there will be no significant dose to the SAWA pump.

In its FIP, the licensee stated that the SAWA flow path inside the reactor building consists of hard pipe 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 Browns Ferry SAWA strategy relies on three instruments: wetwell level; containment pressure; and SAWA flow. Containment pressure and wetwell level are declared Browns Ferry PAM variables as described in RG 1.97 and the existing qualification of these channels is considered acceptable for compliance with Order EA-13-109 in accordance with the guidance in NEI 13-02, Appendix C, Section C.8.1.

In its FIP, the licensee states that the SAWA flow meter is rated for continuous use under the expected ambient conditions and so will be available for the entire period of sustained operation. 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

In its FIP, the licensee stated that calculations of the temperature response of the reactor building and control building during an ELAP event were performed in response to Order EA-12-049. In the severe accident, the core materials are contained inside the primary containment. The temperature response of the reactor building and the control building is driven by the loss of ventilation and ambient conditions and therefore will not change. Thus, the FLEX calculations are acceptable for severe accident use.

For equipment locations outside the reactor building between 7 hours and 7 days when SAWA is being utilized, the licensee performed a qualitative evaluation of equipment and deployment locations and confirmed they are protected by distance and/or buildings with substantial shielding to minimize dose rates. A quantitative evaluation of expected dose rates in AREVA document 51-9262174-003, “Projected Dose Rate Contour Map of Shine from the HCVS Vent Line Extending Above Refueling Floor (BFNP),” was performed per HCVS-WP-02, “Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping,” [Reference 38] and found the dose rates at deployment locations are acceptable.

Table 2 of the FIP provides a list of SAWA/SAWM operator actions as well as an evaluation of each for suitability during a severe accident. Attachments 6 and 6A 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, 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 equipment can all be operated from the MCR or from outside the reactor building at ground level. The Browns Ferry FLEX response ensures that the FLEX pump, FLEX DGs and other equipment can all be run for a sustained period by refueling. All the refueling locations are located in shielded or in areas that are a significant distance from the vent pipe so that there is no radiation hazard from core material during a severe accident. The monitoring instrumentation includes SAWA flow at the FLEX pump, and wetwell level and containment pressure in the MCR.

The NRC staff reviewed the projected environmental conditions prior to the implementation of SAWA earlier in Section 3.1.1.2, “Personnel Habitability – Environmental." The NRC staff also reviewed the information provided and audited the evaluations for projected environmental conditions while implementing, monitoring, and operating the SAWA/SAWM strategy and concludes that the environmental conditions will not prevent operators from performing required actions to implement that plan.

The licensee performed calculation MDQ0000642015000351, “HCVS Operator (Mission) Dose Calculation," which documents the dose assessment for designated areas inside the Browns Ferry reactor building (outside of containment) and outside the Browns Ferry 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 February 11, 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 February 21, 2018 [Reference 17]. The licensee reached its final compliance date on April 10, 2019, and in letter dated June 7, 2019 [Reference 17], has declared that Browns Ferry 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 Browns Ferry to NRC, “Browns Ferry Nuclear Plant, Units 1, 2 and 3 – Phase 1 Overall Integrated Plan in Response to June 6, 2013, Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions Phase 1 (Order Number EA-13-109),” dated June 30, 2014 (ADAMS Accession No. ML14181B169)

3. Letter from Browns Ferry 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 19, 2014 (ADAMS Accession No. ML14353A428)

4. Letter from Browns Ferry 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 29, 2015 (ADAMS Accession No. ML15181A338)

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