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

February 21, 2018

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 - REPORT FOR THE AUDIT OF LICENSEE RESPONSES TO INTERIM STAFF EVALUATIONS OPEN ITEMS RELATED TO NRC ORDER EA-13-109 TO MODIFY LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS (CAC NOS. MF4540, MF4541 AND MF4542; EPID 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 Condition," to all Boiling-Water Reactor licensees with Mark I and Mark II primary containments. The order requirements are provided in Attachment 2 to the order and are divided into two parts to allow for a phased approach to implementation. The order required licensees to submit for review overall integrated plans (OIPs) that describe how compliance with the requirements for both phases of Order EA-13-109 will be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14181B169), Tennessee Valley Authority (TVA, the licensee) submitted its Phase 1 OIP for Browns Ferry Nuclear Plant, Units 1, 2 and 3 (Browns Ferry). By letters dated December 19, 2014, June 29, 2014, December 29, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 30, 2016, December 22, 2016, June 30, 2017, and December 20, 2017 (ADAMS Accession Nos. ML14353A428, ML15181A338, ML15365A554, ML16182A517, ML16357A577, ML17181A333, and ML17354A250, respectively), the licensee submitted its 6-month updates to the OIP. The NRC staff reviewed the information provided by the licensee and issued interim staff evaluations (ISEs) for Phase 1 and Phase 2 of Order EA-13-109 for Browns Ferry by letters dated February 11, 2015 (ADAMS Accession No. ML14356A362), and September 6, 2016 (ADAMS Accession No. ML16244A762), respectively. When developing the ISEs, the staff identified open items where the staff needed additional information to determine whether the licensee's plans would adequately meet the requirements of Order EA-13-109.

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

The NRC staff conducted a teleconference with the licensee on January 25, 2018. The enclosed audit report provides a summary of that aspect of the audit.

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

Sincerely,

RAuluck

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

Docket Nos. 50-259, 50-260 and 50-296

Enclosure:
Audit report

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AUDIT OF LICENSEE RESPONSES TO INTERIM STAFF EVALUATIONS OPEN ITEMS

RELATED TO ORDER EA-13-109 MODIFYING LICENSES

WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF

OPERATION UNDER SEVERE ACCIDENT CONDITIONS

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

BACKGROUND

On June 6, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13143A334), the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-13-109, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Condition," to all Boiling-Water Reactor (BWR) licensees with Mark I and Mark II primary containments. The order requirements are divided into two parts to allow for a phased approach to implementation.

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

Phase 2 of Order EA-13-109 requires license holders of BWRs with Mark I and Mark II primary containments to design and install a system that provides venting capability from the containment drywell under severe accident conditions, or, alternatively, to develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions. The order required all applicable licensees, by December 31, 2015, to submit to the Commission for review an OIP that describes how compliance with the Phase 2 requirements described in Order EA-13-109 Attachment 2 will be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14181B169), Tennessee Valley Authority (TVA, the licensee) submitted its Phase 1 OIP for Browns Ferry Nuclear Plant, Units 1, 2 and 3 (Browns Ferry). By letters dated December 19, 2014, June 29, 2015, December 29, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 30, 2016, December 22, 2016, June 30, 2017, and December 20, 2017 (ADAMS Accession Nos. ML14353A428, ML15181A338, ML15365A554, ML16182A517, ML16357A577, ML17181A333, and ML17354A250, respectively), the licensee submitted its 6-month updates to the OIP, as required by the order.

The NRC staff reviewed the information provided by the licensee and issued interim staff evaluations (ISEs) for Phase 1 and Phase 2 of Order EA-13-109 for Browns Ferry by letters dated February 11, 2015 (ADAMS Accession No. ML14356A362), and September 6, 2016 (ADAMS Accession No. ML16244A762), respectively. When developing the ISEs, the staff identified open items where the staff needed additional information to determine whether the licensee's plans would adequately meet the requirements of Order EA-13-109.

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

AUDIT SUMMARY

As part of the audit, the NRC staff conducted a teleconference with the licensee on January 25, 2017. The purpose of the audit teleconference was to continue the audit review and provide the NRC staff the opportunity to engage with the licensee regarding the closure of open items from the ISEs. As part of the preparation for these audit calls, the staff reviewed the information and/or references noted in the OIP updates to ensure that closure of ISE open items and the HCVS design are consistent with the guidance provided in Nuclear Energy Institute (NEI) 13-02, Revision 1 and related documents (e.g. white papers (ADAMS Accession Nos. ML14126A374, ML14358A040, ML15040A038 and ML15240A072, respectively) and frequently asked questions (FAQs), (ADAMS Accession No. ML15271A148)) that were developed and reviewed as part of overall guidance development. The NRC staff audit members are listed in Table 1. Table 2 is a list of documents reviewed by the staff. Table 3 provides the status of the ISE open item closeout for Browns Ferry. The open items are taken from the Phase 1 and Phase 2 ISEs issued on February 11, 2015, and September 6, 2016, respectively.

FOLLOW UP ACTIVITY

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

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

CONCLUSION

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

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

Table 1 - NRC Staff Audit and Teleconference Participants

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

Table 2 – Audit Documents Reviewed

Caculation MDQ0000642015000351, "Hardened Contatinment System Operator Mission Dose," Revision 1
Calculation MDQ0009992014000291, "Temperatue Response of the Reactor Building Following an Extended Loss of AC Power," Revision 2
Procedure 1-EOI Appendix-13, "Emergency Venting Primary Containment," Revision 3
Procedure 2-EOI Appendix-13, "Emergency Venting Primary Containment," Revision 9
Calculation EDQ0009992013000202, "250 DC Unit Batteries 1, 2, & 3 Evaluation for Beyond Design Basis External Event (BDBEE) Extended Loss of AC Power (ELAP)," Revision 6
Procedure 0-FSI-3F, "Load Shed of 250V Main Bank Battery 1, 2, 3," Revision 1
Calculation NDQ0000642015000341, "HCVS MAAP Analysis," Revision 0
Procedure 0-FSI-4B, "FLEX Communication System Operation," Revision 0
Design Review Report – RAL-70181, Revision 1 – Size 14 Class 150 Wafer Butterfly Valve with Pneumatic Actuator
Report No. S1620.0, Revision 1 – Seismic Test Report for an Optima Stantron Cabinet, Absopulse Voltage Converter, and Moore Industries Signal Converter
Report No S1619.0, Revision 0 – Seismic Test Report for a Panasonic Laptop Computer, Matheson Pressure Gauge, Moore Industries Signal Converter
Seismic Analysis SA-B150912-2, Revision A – ½", ANSI Class 1500 Ball Valve of Stainless Steel Construction Schedule 80 Socket Weld Ends, Lever Operated
Seismic Analysis SA-B150912-1, Revision A – ½", ANSI Class 150 Ball Valve of Stainless Steel Construction Schedule 40 Socket Weld Ends, Lever Operated
Report No. S1615.0, Revision 1 – Seismic Test Report for an Eaton Circuit Breaker Panel
Report No. S1607.0, Revision 1 – Seismic Test Report for an EIZO FlexScan Monitor, Stealth Panel Mount Keyboard, Eaton Disconnect Switches, and Ruhrpumpen/Murphy Controller Panel
Test Report No. PR034998SEI-TR16 – Seismic Qualification Test Report for Yokogawa YS1700 Programmable Indicating Controllers
Seismic Analysis Report for ½ Inch – ANSI Class 150 Shuttle Valve
Design and Seismic Analysis Report for 14 Inch – ANSI Class 150 DRV-Z Nozzle Check Valve
Qualification Summary Report 04518900-QSR, Revision B – HCVS Radiation Monitoring System (DC & AC Input Power Supplies)
Qualification Report 04502054-QR, Revision B – 125 VDC Input Filter and 24V and 125 VDC Input Power Supplies for HCVS RM-1000 System
Qualification Report QR-351025195-1, Revision 2 – Battery Charger
Test Report AZZ PR051230 QP-351025195-1 – Electromagnetic Interference and Suceptibility Tests
Summary of Test Report 06-8680-003, Revision 1 – Nuclear Component Qualification Test Report for the Generic Qualification of Weed Instrument Company Temperature Sensor Assemblies
Calculation NDQ0999910030, "Summary of Mild Environmental Conditions for Browns Ferry Nuclear Plant," Revision 11
Calculation MDQ0009992014000447, "Temperatue Response of the Reactor Building Following a Fukushima Type Severe Accident Utilizing the HCVS," Revision 0
Calculation MDQ0000322015000347, "HCVS Nitrogen Sizing Analysis," Revision 1
Calculation MDQ0003602014000222, "BFN ELAP Transient Analysis," Revision 5

Calculation MDQ0031930018, "BFN Control Bay, Elevation 593.0' and 606.0', and Electric Board Room Analysis," Revision 27
Calculation MDQ0030880213, "Unit 1 and Unit 2 DGB – Central Diesel Information Center Ventilation Requirements," Revision 7
Calculation MDQ0030880208, "U3 DGB Battery Room Ventilation Requirements," Revision 7
Calculation MDN000999201200027, "Thermal Analysis of Control Bay Rooms, Unit 3 Diesel Generator Building Shutdown Board Rooms and Battery #4 Board Room Following Loss of Cooling," Revision 3
Calculation EDQ3030880318, "Electrical Heat Losses – Zones 13, 14, 15, 16," Revision 12
Calculation EDQ3030880319, "Electrical Heat Losses – Zones 11 and 12," Revision 9
Calculation EDQ0030910058, "Electrical Equipment Heat Losses – Individual Rooms in Units 1, 2 and 3," Revision 45
Calculation EDQ0010642015000349, "Unit 1 HCVS Electrical Design & Equipment Sizing Analysis," Revision 2
Calculation EDQ0000642016000510, "Unit 2 & 3 HCVS DC Electrical Design & Equipment Sizing Analysis," Revision 0
Design Change Technical Evaluation – DCN 71391 (Unit 3)
Design Change Technical Evaluation – DCN 71390 (Unit 2)
Design Change Technical Evaluation – DCN 71389 (Unit 1)
BFN White Paper R06 160315 491 – Validation of NEI White Paper HCVS-WP-04 First Assumption for Missile Protection of Hardened Containment Vent System at Browns Ferry Nuclear Plant
Calculation MDN0003602014000233, "Hydraulic Analysis for Fukushima FLEX Connection Modifications," Revision 3
Calculation MDQ0000642015000393, "HCVS Equipment Dose Evaluation," Revision 2
AREVA Document 51-9262174-003 – Projected Dose Rate Contour Map of Shine from the HCVS Vent Line Extending Above Refueling Floor (BFNP)
BWROG-TP-008, "Severe Accident Water Addition Timing"
BWROG-TP-011, "Severe Accident Water Management Supporting Evaluations"

Browns Ferry Nuclear Plant, Units 1, 2 and 3
Vent Order Interim Staff Evaluation Open Items:

Table 3 - ISE Open Item Status Table

ISE Open Item Number Requested Action	Licensee Response – Information provided in 6 month updates and on the ePortal	NRC Staff Close-out notes	Safety Evaluation (SE) status Closed; Pending; Open (need additional information from licensee)
Phase 1 ISE OI 1 Make available for NRC staff audit an evaluation of temperature and radiological conditions to ensure that operating personnel can safely access and operate controls and support equipment.	An evaluation of temperature and radiological conditions was performed to ensure that Operating personnel can safely access and operate controls at the Remote Operating Station located in the Diesel Buildings and in the Reactor building. This evaluation is documented in Unit 1 [Design Change] DCN 71389 Design Change Technical Evaluation (Page 70-73 of 81), Unit 2 DCN 71390 Design Change Technical Evaluation (Page 65-68 of 75), and Unit 3 DCN 71391 Design Change Technical Evaluation (Page 60-63 of 69). MDQ0000642015000351 "HCVS OPERATOR (MISSION) DOSE CALCULATION", and MDQ0009992014000291 "TEMPERATURE RESPONSE OF THE REACTOR BUILDING FOLLOWING AN EXTENDED LOSS OF AC POWER" were used to validate the evaluation.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. MDQ0003602014000222, "BFN ELAP Transient Temperature Analysis" shows with compensatory actions the main control rooms (MCRs) for U-1, U-2, & U-3 remain below 110°F at 72 hours. MDQ0030880213, "Unit 1 and Unit 2 DGB - Central Diesel Information Center Ventilation Requirements," determines ventilation requirements for the remote operation station (ROS) area during normal operation. This calculation is for determining normal ventilation requirements. 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 in the area of the ROS along with no major	Closed [Staff evaluation to be included in SE Sections 3.1.1.2 and 3.1.1.3]

		electrical heat loads or residual heat loads from operating equipment, area temperatures will not be adverse to operators performing their required actions in the ROS. MDN0009992012000027, "Thermal Analysis of Control Rooms, Unit 3 DGB SDBRs and BR4BR Following Loss of Cooling," evaluated the Unit-3 ROS in the Unit-3 DG [diesel generator] Shutdown Board Room (SDBR). Temperature remains below 110°F. MDQ0000642015000351, "HCVS Operator (Mission) Dose Calculation," was performed to determine the integrated radiation dose due to HCVS operation. Temperature and radiological conditions should not inhibit operator actions needed to initiate and operate the HCVS during an ELAP with severe accident conditions. No follow-up questions.	
Phase 1 ISE OI 2 Make available for NRC audit documentation that procedure 1/2/3-EOI Appendix 13 has been revised to include venting for loss of dc power.	1-EOI-Appendix 13 revised to include venting for loss of DC [direct current] power. Revision 3 issued on 10/19/2016. 2-EOI-Appendix 13 revised to include venting for loss of DC power. Revision 9 issued on 3/24/2017.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The guidelines and procedures for HCVS operation are complete for Units 1 & 2 and consistent with the guidance in NEI 13-02.	Closed [Staff evaluation to be included in SE Section 5.1]

	3-EOI-Appendix 13 will be revised following completion of the installation of Unit 3 Hardened Containment Vent System.	The Unit 3 procedure for HCVS operation will be revised following completion of the installation of Unit 3 HCVS and will follow the same guidance as the other two units, consistent with the guidance in NEI 13-02. No follow-up questions.	
Phase 1 ISE OI 3 Make available for NRC staff audit documentation demonstrating that all load sheds will be accomplished within one hour of event initiation and will occur in an area not impacted by a possible radiological event.	Calculation EDQ0009992013000202, 250V DC Unit Batteries, 1, 2, & 3 Evaluation for the Beyond Design Basis External Event (BDBEE) Extended Loss of AC Power (ELAP), has been issued to determine load shedding impact on the unit batteries. The performance of the load shed is directed by 0-FSI-1 (Page 1) "FLEX Support Instruction" and performed in accordance with 0-FSI-3F "Load Shed of 250V Main Bank Battery 1,2,3." The load shed is performed in the Control Bay and Electrical Board rooms only and will not require entry into areas that are impacted by a possible radiological event.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. EDQ0009992013000202, "250V DC Unit Batteries, 1, 2, & 3 Evaluation for the Beyond Design Basis External Event (BDBEE) Extended Loss of AC Power (ELAP)," Rev. 6. Coping time of Unit batteries extended from 8 hours to 12 hours. FLEX procedure 0-FSI-3F, "Load Shed of 250V Main Bank Battery 1, 2, 3," Rev. 1. The load shed is performed in the Control Bay and Electrical Board rooms and will not require entry into areas that are impacted by a possible radiological event. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.6]
Phase 1 ISE OI 4 Make available for NRC staff audit documentation that demonstrates that operating units that have not	A conceptual meeting was held in November 2014, and a staging plan was used to separate the existing HWWV from the HCVS. The Hardened Containment Vent System has been implemented on Unit 1 and Unit 2. Unit 3 remains capable	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal.	Closed [Staff evaluation to be included in SE Section 3.1.2.3]

implemented the order will be able to vent through the existing vent system unaffected by the implementation of HCVS on other units.	of using the existing HWWV system until completion of the Hardened Containment Vent System which will be performed in the Spring of 2018.	Units are not interconnected. There are no cross-ties between units. No follow-up questions.	
Phase 1 ISE OI 5 Make available for NRC staff audit analyses demonstrating that HCVS has the capacity to vent the steam/energy equivalent of one percent of licensed/rated thermal power (unless a lower value is justified), and that the suppression pool and the HCVS together are able to absorb and reject decay heat, such that following a reactor shutdown from full power containment pressure is restored and then maintained below the primary containment design pressure and the primary containment pressure limit.	The existing wetwell vent and the HCVS have been designed for 1 percent of rated thermal power at extended power uprate (EPU) (3952 MWt) conditions. This analysis is available and documented in Calculation NDQ0000642015000341.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. NDQ0000642015000341, "HCVS MAAP Analysis" uses the Computer Code MAAP, Version 4.0.7 to confirm the vent design. The HCVS can be opened at 56 pounds per square inch gauge (psig) and closed at a lower pressure. Based on the preliminary study, a vent size of 14" Schedule 40 was chosen. At 70.7 per square inch absolute (psia) (56psig), the corresponding required rate of flow is: 1% · Rated Power /(hg(@70.7psia) = 0.01·3952MW (947.817BTU /(s. mega watt(MW)) (lbm/1101BTU) = 31.7 lbm/sec Upon examination of the GOTHIC analysis (Reference [12]), it can be seen that GOTHIC calculates the HCVS flow rate capability at this pressure (70.7 psia, design pressure of the drywell and wetwell vent conditions) to be 58.4 lbm/sec. Since the	Closed [Staff evaluation to be included in SE Section 3.1.2.1]

		calculation satisfying the regulatory requirement set forth in EA 13-109 (1 % decay heat removal capability requirement) requires a minimum of 31.7 lbm/sec, the current design of the HCVS safely satisfies this requirement. No follow-up questions.	
Phase 1 ISE OI 6 Make available for NRC staff audit documentation that demonstrates adequate communication between the remote HCVS operation locations and HCVS decision makers during ELAP and severe accident conditions.	A communication system has been implemented (DCN 70852) that uses hand held radios for communication between the MCR and the ROS. This Radio System consists of a UHF/VHF trunked system and an independent VHF channel (F4). The In-plant Radio System is accessed by handheld radios. The In-plant Radio System has normal and emergency diesel generator backed power supply. The radio system is powered from two Class 1E redundant power sources, 480V DG Auxiliary Boards A and B. Primary power source will be from 480V DG Auxiliary Board A via second 480-208/120 transformer / distribution center. In the event of loss of primary power source, power to radio equipment will be automatically transferred to backup source via transfer switches located in each cabinet, with exception of cabinet 4, which receives power via cabinet 1 transfer switch. Backup power source includes UPS with battery capacity to supply four (4) UHF channels for three hours. Therefore in this configuration capacity is reduced from five simultaneous conservations to three. The loads supplied via UPS can be	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The communication methods are the same as accepted in Order EA-12-049. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.1.1]

	alternatively supplied from portable generator via transfer switch (0-FSI-4B). UPS Conservation can be accomplished by switching off one of the two UPSs until such time the active UPS reaches "low level". Then the UPS previously switched off can be returned to service extending the overall time the radio system can remain operable without portable generator power to approximately 6 hours. BFN maintains a large number of handheld radios, batteries and charging units. The FLEX program does not maintain dedicated handheld radios. These units, spare batteries and chargers will be gathered if not readily available in the control rooms. Handheld Radios can additionally be operated in "Radio-to-Radio" mode enabling communications not affected by shielding or distance.		
Phase 1 ISE OI 7 Make available for NRC staff audit documentation of an evaluation verifying the existing containment isolation valves, relied upon for the HCVS, will open under the maximum expected differential pressure during BDBEE and severe accident wetwell venting.	An evaluation was performed 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 Rev. 1.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. RAL-70181 Rev 1 Design Review Report "Size 14 Class 150 Wafer Butterfly Valve with Pneumatic Actuator Drawing: 94-15972," shows a maximum stem torque is 6000 in-lbf. Maximum pressure differential = 70.7 psi. The operating torque at the seat is expected to increase approximately 11 % due to the	Closed [Staff evaluation to be included in SE Section 3.2.1]

		increase in differential pressure from 56 psi to 70.7 psi. Using current methods and parameters the calculated required torque to start open increases from 4944 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 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. No follow-up questions.	
Phase 1 ISE OI 8 Make available for NRC staff audit documentation of a seismic qualification evaluation of HCVS components.	Seismic/Qualification Reports for Units 1, 2, & 3 HCVS Components tables, file HCVS Phase 1 ISE OI-8, and documents are attached. From the tables, use the Seismic Report # column to find the reports.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee provided several reports which demonstrate the seismic adequacy of the HCVS components required for HCVS venting remain functional following a design-basis earthquake. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.2.2]
Phase 1 ISE OI 9 Make available for NRC staff audit descriptions of all	Seismic/Qualification Reports for Units 1, 2, & 3 HCVS Components tables, file HCVS Phase 1 ISE OI-9. From the	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal.	Closed

instrumentation and controls (existing and planned) necessary to implement this order including qualification methods.	tables, use the Qualification Report # column to find the reports.	The existing plant instuments required for HCVS (i.e. wetwell level instruments and drywell pressure instruments) meet the requirements of Regulatory Guide (RG) 1.97. The HCVS components tables and associated qualification reports provide the qualifications for new HCVS I&C components. The staff's review indicated that the qualification met the order requirements. No follow-up questions.	[Staff evaluation to be included in SE Section 3.1.2.8]
Phase 1 ISE OI 10 Make available for NRC staff audit the descriptions of local conditions (temperature, radiation and humidity) anticipated during ELAP and severe accident for the components (valves, instrumentation, sensors, transmitters, indicators, electronics, control devices, and etc.) required for HCVS venting including confirmation that the components are capable of performing their functions during ELAP and severe accident conditions.	The DCN determined that only three components per unit of the HCVS would experience harsh post-event environmental conditions (high temperatures and humidity). These three components were procured with the requirements specified and the vendors provided qualification reports documenting the capabilities of each component. BFN HCVS components subjected to post event harsh environments were specified and procured with capabilities exceeding those requirements.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The DCN discusses the environmental conditions during an accident at the locations containing instrumentation and controls (I&C) components. The staff's review indicated that the environmental qualification met the order requirements. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.1.4]
Phase 1 ISE OI 11 Make available for NRC staff audit the final sizing evaluation for HCVS batteries/battery	From DCN 71389 (Unit 1) Design Change Technical Evaluation (Page 36 of 81): BFN Calculation EDQ0010642015000349 was performed to size and perform electrical analysis on the HCVS batteries	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal.	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

charger including incorporation into FLEX DG loading calculation.	and battery charger. BFN Calculation EDQ0010642015000349 also performs electrical analysis on the new 250 VDC [volt direct current] power Distribution Panel (1-BDDD-064-0001), the new 250V DC to DC Converter (1-CNV-064-0001), and determines parameters for isolation diodes 1-DIO-064-221A, 1-DIO-064-221B, 1-DIO-064-222A, and 1-DIO-064-222B. Battery sizing was performed in accordance with IEEE [Institute of Electrical and Electronics Engineers] Std. 485-2010 and electrical equipment sized in accordance with IEEE Std. 946-2004. From DCN 71390 (Unit 2) Design Change Technical Evaluation (Page 35 of 75): BFN Calculation EDQ0000642016000510 was performed to size and perform electrical analysis on the HCVS batteries and battery charger. BFN Calculation EDQ0000642016000510 also performs electrical analysis on the new 250 VDC power Distribution Panel (O-BDDD-064-0001), the new 250V DC to DC Converter (2-CNV-064-0001), and determines parameters for isolation diodes (2-DI0-064-221A, 2-010-064-221 B, 2-DI0-064-222A, and 2-DI0-064-222B). Battery sizing was performed in accordance with IEEE Std. 485-2010 and electrical equipment sized in accordance with IEEE Std. 946-2004. Note the HCVS battery and battery charger installed by this DCN have been sized to accommodate both Unit 2 and Unit 3 HCVS electrical loads. From DCN 71391 (Unit 3) Design Change Technical Evaluation (Page 32 of 69):	The licensee stated that all electrical power required for operation of HCVS components is provided by the 250 VDC battery/battery charger. The battery sizing calculations (EDQ0010642015000349 - Unit 1 & EDQ0000642016000510 - Unit 2 and 3) confirmed that the HCVS batteries have a minimum capacity capable of providing power for 24 hours without recharging, and therefore is adequate. There is no incorporation required into the FLEX DG loading calculation for any Unit due to there are no plans or requirements to recharge the HCVS battery after depletion. The HCVS electrical loads would be aligned back to their normal power supply which is the Unit Battery. No follow-up questions.	

	BFN Calculation ED00000642016000510 was performed to size and perform electrical analysis on the HCVS batteries and battery charger. BFN Calculation EDQ000642016000510 also performs electrical analysis on the new 250 VDC power Distribution Panel (O-BDDD-064-0001), the new 250V DC to DC Converter (3-CNV-064-0001), and determines parameters for isolation diodes (3-DI0-064-221A, 2-010-064-221 B, 3-DI0-064-222A, and 3-DI0-064-222B). Battery sizing was performed in accordance with IEEE Std. 485-2010 and electrical equipment sized in accordance with IEEE Std. 946-2004. Note the HCVS battery and battery charger installed by this DCN have been sized to accommodate both Unit 2 and Unit 3 HCVS electrical loads. There is no incorporation required into the FLEX DG loading calculation for any Unit due to there are no plans or requirements to recharge the HCVS battery after depletion. The HCVS electrical loads would be aligned back to their normal power supply which is the Unit Battery.		
Phase 1 ISE OI 12 Make available for NRC staff audit documentation of the HCVS nitrogen pneumatic system design including sizing and location.	Evaluation has been completed and documented in DCN 71389 for Unit 1, DCN 71390 for Unit 2, DCN 71391 for Unit 3 and calculation MDQ0000322015000347 "HCVS NITROGEN SIZING ANALYSIS". As documented in DCN 71389 Design Change Technical Evaluation (Page 29 of 81) there are 9 Nitrogen Cylinders required for Unit 1 for 7 days of Hardened Vent operation. There are 5 Nitrogen Cylinders installed to support Hardened	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The DCN and calculation MDQ0000322015000347 discusses the pneumatic design and sizing. DCN 71389, DCN 71390, and DCN 71391 discusses the required number of nitrogen cylinders needed for vent	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

	Vent operation for Unit 1. As documented in DCN 71390 Design Change Technical Evaluation (Page 28 of 75) there are 10 Nitrogen Cylinders required for Unit 2 for 7 days of Hardened Vent operation. As documented in DCN 71391 Design Change Technical Evaluation (Page 26 of 69) there are 9 Nitrogen Cylinders required for Unit 3 for 7 days of Hardened Vent operation. (24 Nitrogen Cylinders are required for Unit 2 and 3 for 7 days simultaneous operation. There are 5 Nitrogen Cylinders installed to support Hardened Vent operation for Unit 2 and 3. There are 6 Nitrogen Cylinder carts with 6 Nitrogen Cylinders on each cart available in the FLEX Storage building with no other committed use of them.	operation for sustained operation for each unit, respectively. The number of nitrogen cylinders installed in each unit and available are sufficient to operate the HCVS for 24 hours. No follow-up questions.	
Phase 1 ISE OI 13 Make available for NRC staff audit the seismic and tornado missile final design criteria for the HCVS stack.	Tornado and seismic missile criteria are located in System 64A Design Control Document (DCD). As part of DCN 71389 (Unit 1) and DCN 71390 (Unit 2) the DCD was revised. For DCN 71391 (Unit 3) a markup reflecting these changes has been generated and will be incorporated in the next revision of the DCD per TVA process. As stated in the OIP the HCVS design provides missile protection from ground level to 30 feet in accordance with NRC Regulatory Guide 1.76 based on a site-specific tornado missile evaluation. Above 30 feet the exposed vent piping will be robustly designed in accordance with HCVS-WP-04. This is a design consideration using reasonable protection features for the screened in hazards from NEI 12-06. (reference HCVS-FAQ-04; HCVS-WP-04). Browns Ferry has utilized	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. DCN 71389, DCN 71390, and DCN 71391 addresses the HCVS seismic qualification and tornado missile design for each unit, respectively. . The licensee evaluated the entire HCVS system to Seismic Catergory I, which is consistent with the plants seismic design-basis. For the tornado missile design, the licensee's design is consistent with the endorsed white paper HCVS-WP-04 and meets all of the applicable tornado missile	Closed [Staff evaluation to be included in SE Section 3.2.2]

	NEI HCVS-WP-04 "Missile Evaluation for HCVS Components 30 Feet Above Grade" and meet the assumptions in NEI HCVS-WP-04 as stated below: 1. Piping and components external to any missile-protected structure and less than 30 feet above grade are evaluated and, unless otherwise justified in plant-specific OIPs, protected from large and small missiles. How met: The various BFN site areas were reviewed for their potential to create missiles, defined by NRC Regulatory Guide 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 BFN White Paper "Validation of NEI White Paper HCVS-WP-04 First Assumption for Missile Protection of Hardened Containment Vent System at Browns Ferry Nuclear Plant" 2. Piping and components external to any missile-protected structure and greater than 30 feet above grade conform to the following: a. The target area of the HCVS components is less than 300 ft^2,	assumptions identified in the white paper. No follow-up questions.	

	How met: As stated in DCN 71389 (Unit 1) DCN 71390 (Unit 2) and DCN 71391 (Unit 3) DCD 7064A Mark-up the HCVS pipe is 14 inch diameter and has non-protected pipe runs not in excess of 250' which results in a potential missile target area less than the 300 ft^2 limit specified in HCVS-WP-04 b. The size and robustness of the exposed HCVS s are substantial (e.g., steel piping versus small tubing or plastic piping), How met: As stated in DCN 71389 (Unit 1) DCN 71390 (Unit 2) and DCN 71391 (Unit 3) DCD 7064A Mark-up the HCVS piping is constructed of 14" schedule 40 carbon steel piping providing substantial robustness, thereby satisfying HCVS-WP-04. c. There is no source of obvious potential missiles in the proximity of the exposed HCVS components (such as an unrestrained material lay down area). How met:The Reactor Building roof contains a limited amount of rigidly mounted permanent plant equipment with non-anchored temporary equipment being stored on the roof at a low frequency. Therefore it is not credible that an item located on the roof of the Reactor Building will cause loss of function of the HCVS piping. Signs have been posted on the south side of the Reactor Building in area of exposed HCVS piping to prohibit the storage or placing of equipment within the		

	proximity of the exposed HCVS piping and this restriction is documented in 0-TPP-ENG-632(Bases) Diverse and Flexible Coping Strategies (FLEX) Program Bases Document. 3. Licensees consider guidance in FLEX, or other procedures, to restore venting capability in the event the HCVS is damaged. Restoration could include cutting pipe below damaged section. This location may have to be below the release height requirements otherwise imposed. How met: As stated in DCN 71389 (Unit 1) DCN 71390 (Unit 2) and DCN 71391 (Unit 3) DCD 7064A Mark-up appropriate assessment and restoration of venting capacity in the event the HCVS is damage is conducted under procedure O-FSI-6A "Damage Assessment." The HCVS piping was designed to provide a Tee in the piping in the event that the piping is damaged and will require cutting. 4. Licensees verify that if hurricanes are screened in for FLEX (see NEI-12-06), that the site procedures recommend a plant shut down prior to hurricane arrival on-site. As stated in DCN 71389 (Unit 1) DCN 71390 (Unit 2) and DCN 71391 (Unit 3) DCD 7064A Mark-up in accordance with NEI 12-06, BFN is not susceptible to winds exceeding 130 mph from hurricanes. As such, Browns Ferry Nuclear Plant screens out for hurricanes and HCVS-WP-04 assumption 4 does not apply.		

Phase 1 ISE OI 14 Provide a description of the final design of the HCVS to address hydrogen detonation and deflagration.	A description of the final design of the HCVS to address hydrogen detonation and deflagration is contained in the Design Change Technical Evaluation (section 6.4.8) of Unit 1 DCN 71389, Unit 2 DCN 71390, and Unit 3 DCN 71391.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee's design concept for use of the check valve is that after venting, steam and Hydrogen/CO are isolated in the pipe volume between the downstream check valve and the upstream primary containment isolation valve. As the vented steam cools a vacuum forms in the HCVS piping. The check valve prevents this vacuum from pulling oxygen into the vent pipe and creating a combustible mixture. For this approach check valve 1(2,3)-CKV-064-0802 is located less than the run-up distance to DDT which is determined by 30 LID (NEI 13-02, Ref 4.1.6). Based on the internal pipe diameter of 14" schedule 40 piping, the check valve must be located within 35 feet of the vent opening which is located at the 741 '-6" elevation. The location of this check valve is at approximately elevation 722' just above the Unit 1(2,3) Reactor Building roof, approximately 20 feet from the vent opening, which is less than the minimum run-up distance to DDT. The design also addressed the potential for oxygen entering a condensate drain downstream of check valve 1(2,3)-FCV-064-022.	Closed [Staff evaluation to be included in SE Section 3.1.2.11]

		The drain line has two check valves in series which will greatly restrict in leakage of oxygen ECP 71389, 71390, and 71391 indicates that HCVS piping for each unit is routed independently up the southern exterior wall of the Reactor Building. The licensee's design is consistent with Option 5 of the endorsed white paper HCVS-WP-03. No follow-up questions.	
Phase 1 ISE OI 15 Provide a description of the strategies for hydrogen control that minimizes the potential for hydrogen gas migration and ingress into the reactor building or other buildings.	A description of the strategies for hydrogen control that minimizes the potential for hydrogen gas migration and ingress into the reactor building or other buildings is contained in the Design Change Technical Evaluation (section 6.4.8) of Unit 1 DCN 71389, Unit 2 DCN 71390, and Unit 3 DCN 71391.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. ECP 71389, 71390, and 71391 indicate that HCVS piping for each unit is routed independently up the southern exterior wall of the Reactor Building. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.12]
Phase 1 ISE OI 16 Provide design details that minimize unintended cross flow of vented fluids within a unit and between units on the site.	A description of the design details that minimize unintended cross flow of vented fluids within a unit and between units on the site is contained in the Design Change Technical Evaluation (section 6.4.10) of Unit 1 DCN 71389, Unit 2 DCN 71390, and Unit 3 DCN 71391.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee's design appears to minimize the unintended cross flow of vented fluids. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.3]
Phase 2 ISE OI 1	Hydraulic Analysis calculation MDN0003602014000233 was revised to include a bounding case that concluded	The NRC staff reviewed the information provided in the 6-	Closed

Licensee to perform a hydraulic evaluation to ensure flow adequacy can be met for all 3 units using 1 FLEX pump to support SAWA flow requirement.	that a single FLEX pump (with booster pump) can provide 500 gpm [gallons per minute] to Unit 1 RPV, 500 gpm to Unit 2 RPV, and 500 gpm to Unit 3 RPV (each at RPV pressure of 106 psig) in response to a SAWA [severe accident water addition] event.	month updates and on the ePortal. The hydraulic analysis, "MDN0003602014000233" shows that the required SAWA flowrate of 500 gpm for each unit, is within the capacity of the single FLEX pump (with booster pump). No follow-up questions.	[Staff evaluation to be included in SE Section 4.1.1.2]
Phase 2 ISE OI 2 Licensee to evaluate the SAWA [severe accident water addition] equipment and controls, as well as ingress and egress paths for the expected severe accident conditions (temperature, humidity, radiation) for the sustained operating period.	<u>Equipment and Controls</u> Plant instrumentation for SAWM [severe accident water management] that is qualified to RG 1.97 or equivalent is considered qualified for the sustained operating period without further evaluation. The following plant instruments are qualified to RG 1.97: DW Pressure 1,2,3-PI-64-67B Suppression Pool Level 1,2,3-LI-64-159A Passive components that do not need to change state after initially establishing SAWA flow do not require evaluation beyond the first 8 hours, at which time they are expected to be installed and ready for use to support SAWA/SAWM. The following additional equipment performing an active SAWA/SAWM function is considered: SAWA/SAWM flow instrument. SAWA/SAWM pump	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. AREVA document 51-9262174-003 shows that radiological conditions should not inhibit operator actions or SAWA equipment and controls needed to initiate and operate the HCVS during an ELAP with severe accident conditions. The temperature evaluation addressed in Phase 1 Open Item #1 bounds the SAWA/SAWM operation. No follow-up questions.	Closed [Staff evaluation to be included in SE Sections 4.5.1.2 and 4.5.1.3]

	FLEX generator SAWA Throttle valve These components will be used at a remote location (outside reactor building) and have been evaluated for the environmental conditions applicable at those locations. <u>Ingress and Egress</u> A quantitative evaluation of expected dose rates AREVA document 51-9262174-003 "Projected Dose Rate Contour Map of Shine from the HCVS Vent Line Extending Above Refueling Floor (BFNP)" has been performed per HCVS-WP-02 and found the dose rates at deployment locations including ingress/egress paths are acceptable.		
Phase 2 ISE OI 3 Licensee to demonstrate how SAWA flow is capable to perform its intended function for the sustained operating period under the expected temperature and radiological conditions.	The SAWA throttle valve will be used at a remote location (outside reactor building) and has been evaluated for the environmental conditions applicable at that location. A validation was performed on the SAWA flowmeter that determined thru discussion with the vendor (Fire Research Corporation) that the ambient temperature limits of the flowmeter are a minimum of -20 degrees F and a maximum temperature of 150 degrees F when an external power source is being used. Browns Ferry Nuclear Plant has a minimum ambient extreme daily temperature record of -12 degrees F and a maximum ambient extreme daily temperature of 108 degrees F as	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. AREVA document 51-9262174-003 shows that radiological conditions should not inhibit the operation of the SAWA throttle valve during an ELAP with severe accident conditions. The temperature evaluation addressed in Phase 1 Open Item #1 bounds the location of the SAWA throttle valve. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 4.5.1.2]

	documented in the EA-12-049 Overall Integrated Plan. The BFN Flowmeter is using external batteries (FLEX pump diesel) to power flowmeter which allow operations within the extreme daily temperatures of -12 degrees F and 108 Degrees F.		
Phase 2 ISE OI 4 Make available for NRC staff supporting documentation demonstrating that containment failure as a result of overpressure can be prevented without a drywell vent during severe accident conditions.	The wetwell vent has been designed and installed to meet NEI 13-02 Rev 1 guidance which will ensure that it is adequately sized to prevent containment overpressure under severe accident conditions. The SAWM strategy will ensure that the wetwell vent remains functional for the period of sustained operation. Browns Ferry Nuclear Plant will follow the guidance (flow rate and timing) for SAWA/SAWM described in [Boiling-Water Reactors Owners Group] BWROG-TP-15-008 and BWROG-TP-15-011. These documents have been posted for NRC staff review. The wetwell vent will be opened prior to exceeding the PCPL [Primary Containment Pressure Limit] value of 62 PSIG. Therefore, containment over pressurization is prevented without the need for a drywell vent.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. BWROG-TP-15-008 demonstrates adding water to the reactor vessel within 8-hours of the onset of the event will limit the peak containment drywell temperature significantly reducing the possibility of containment failure due to temperature. Drywell pressure can be controlled by venting the suppression chamber through the suppression pool. BWROG-TP-15-011 demonstrates that starting water addition at a high rate of flow and throttling after approximately 4-hours will not increase the suppression pool level to that which could block the suppression chamber HCVS. As noted under Phase 1, the vent is sized to pass a minimum steam flow equivalent to 1% rated thermal power. This is sufficient to permit venting to maintain	Closed [Staff evaluation to be included in SE Section 4.2]

		containment below the lower of PCPL or design pressure. No follow-up questions.	
Phase 2 ISE OI 5 Make available for the NRC staff a description of how the plant is bounded by the reference plant analysis that shows the SAWM strategy is successful in making it unlikely that a drywell vent is needed.	<u>Reference Plant</u> Torus freeboard volume is 525,000 gallons, SAWA flow is 500 GPM [gallons per minute] at 8 hours followed by 100 GPM from 12 hours to 168 hours, <u>Browns Ferry</u> Torus freeboard volume is 757,544 gallons, SAWA flow is 500 GPM at 8 hours followed by 100 GPM from 12 hours to 168 hours. The above parameters for Browns Ferry compared to the reference plant that determine success of the SAWM strategy demonstrate that the reference plant values are bounding. Therefore, the SAWM strategy implemented at Browns Ferry makes it unlikely that a drywell vent is needed to prevent containment overpressure related failure. Note: Determined Torus Freeboard volume by using values from 0-TI-394 and subtracting the air volume above the Vent line (26.3 Ft) from the air volume from the top of the normal Suppression pool water level (15.08 Ft). Using values from 0-TI-394, the air volume above 26.3 is 192765.2 gallons. The air volume at -1" (15.08) is 950,309.8. 950,309.8 - 192765.2 = 757544.6	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The staff reviewed the parameters from the reference plant to those of Browns Ferry. The staff concurs that it is unlikely the suppression chamber HCVS could become blocked leading to a successful SAWA/SAWM strategy. Therefore, it is unlikely a drywell vent would be required to maintain containment integrity. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 4.2.1.1]

Phase 2 ISE OI 6 Make available for NRC staff documentation that demonstrates adequate communication between the MCR and the operator at the FLEX pump during severe accident conditions.	Browns Ferry Nuclear Plant utilizes the Harris Radio System to communicate between the MCR and the operator at the FLEX pump. This communication method is the same as accepted in Order EA-12-049. These items will be powered and remained powered using the same methods as evaluated under Order EA-12-049 and continued for the period of sustained operation.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The communication methods are the same as accepted in Order EA-12-049. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 4.1]

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3 - REPORT FOR THE AUDIT OF LICENSEE RESPONSES TO INTERIM STAFF EVALUATIONS OPEN ITEMS RELATED TO NRC ORDER EA-13-109 TO MODIFY LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS (CAC NOS. MF4540, MF4541 AND MF4542; EPID L-2014-JLD-0044) DATED February 21, 2018

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