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

March 22, 2018

Mr. William R. Gideon
Site Vice President
Brunswick Steam Electric Plant
8470 River Rd., SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - 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. MF4467 AND MF4468; EPID L-2014-JLD-0041)

Dear Mr. Gideon:

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

By letter dated June 26, 2014 (ADAMS Accession No. ML14191A687), Duke Energy Progress, LLC (Duke, the licensee) submitted its Phase 1 OIP for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP, Brunswick). By letters dated December 17, 2014, June 25, 2015, December 11, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 28, 2016, December 15, 2016, June 19, 2017, and December 20, 2017 (ADAMS Accession Nos. ML14364A029, ML15196A035, ML16020A064, ML16190A111, ML16365A007, ML17171A383, and ML17354A248, 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 Brunswick by letters dated March 10, 2015 (ADAMS Accession No. ML15049A266), and August 17, 2016 (ADAMS Accession No. ML16223A725), 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 February 22, 2018. The enclosed audit report provides a summary of that aspect of the audit.

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

Sincerely,

RAuluck

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

Docket Nos. 50-325 and 50-324

Enclosure:
Audit report

cc w/encl: Distribution via Listserv

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

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

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

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 26, 2014 (ADAMS Accession No. ML14191A687), Duke Energy Progress, LLC (Duke, the licensee) submitted its Phase 1 OIP for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP, Brunswick). By letters dated December 17, 2014, June 25, 2015, December 11, 2015 (which included the combined Phase 1 and Phase 2 OIP), June 28, 2016, December 15, 2016, June 19, 2017, and December 20, 2017 (ADAMS Accession Nos. ML14364A029, ML15196A035, ML16020A064, ML16190A111, ML16365A007, ML17171A383, and ML17354A248, 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 Brunswick by letters dated March 10, 2015 (ADAMS Accession No. ML15049A266), and August 17, 2016 (ADAMS Accession No. ML16223A725), 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 February 22, 2018. The purpose of the audit teleconference was to continue the audit review and provide the NRC staff the opportunity to engage with the licensee regarding the closure of open items from the ISEs. As part of the preparation for 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 Brunswick. The open items are taken from the Phase 1 and Phase 2 ISEs issued on March 10, 2015, and August 17, 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 Brunswick, 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, but notes that some open items are designated by the staff to be open or pending as described in Table 3 below. The status of the NRC staff's review of these open items may change as additional information is provided to the staff, or 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

Procedure 0PLP-01.4, "Fukishima FLEX System Availability, Action, and Surveillance Requirements," Revision 6
Calculation 0FLEX-0035, "Flow Capacity of BNP Hardened Wetwell Vent Units 1 & 2 at 1% Rated Power," Revision 0
Calculation BNP-MECH-FLEX-002, "Brunswick Nuclear Plant Containment Analysis of FLEX Strategies," Revision 0
Calculation 31116-CALC-E-001, "FLEX Diesel Generator Sizing Caculation," Revision 0
EC 289233, "Fukushima: Hardened Vents at BNP," Revision 2
Procedure 0EOP-01-FSG-04, "FLEX Diesel Generator Alignment," Revision 3
Calculation BNP-MECH-FLEX-005, "Brunswick Nuclear Plant MAAP 5.02 Analysis to Support SAWA Strategy," Revision 1
Calculation BNP-MECH-FLEX-004, "BNP Hardened Wetwell Check Valve Air Inleakage Evaluation," Revision 0
Calculation BNP-MECH-FLEX-003, "FLEX Control Building GOTHIC Heatup Analysis," Revision 0
Calculation BNP-MECH-FLEX-001, "FLEX Reactor Building GOTHIC Heatup Analysis," Revision 0
Calculation BNP-MECH-AOV-DP-CAC, "Differential Pressure Calculations for ½ -CAC-V7-AO, -V8-AO, -V216-AO – Inboard Suppression Pool Purge Exhaust, Outboard Suppression Pool Purge Exhaust, and Hardened Wetwell Vent Isolation Air-Operated Valves," Revision 0
Calculation BNP-MECH-2-CAC-V216-AO, "Hardened Wetwell Vent Outboard Isolation Valve," Revision 1
Calculation BNP-MECH-2-CAC-V7-AO, "Torus Purge Exhaust Valve," Revision 1
Calculation BNP-MECH-2-CAC-V216-AO, "AOV Setup Calculation for 1-CAC-V216-AO Hardened Wetwell Vent Outboard Isolation Valve," Revision 1
Calculation BNP-MECH-2-CAC-V7-AO, "AOV Setup Calculation for 1-CAC-V7-AO Torus Purge Exhaust Valve," Revision 1
Calculation BNP-E-6.125, "24/48 VDC Battery Allowable Discharge Rate for HCVS ELAP," Revision 0
Calculation 0RNA-001, "Instrument Air Nitrogen Backup System Volume Requirements," Revision 4
Calculation 0FLEX-0003, "Hydraulic Analysis for Fukushima FLEX Connection Modifications," Revision 2
BWROG-TP-008, "Severe Accident Water Addition Timing"
BWROG-TP-011, "Severe Accident Water Management Supporting Evaluations"

Brunswick Steam Electric Plant, Units 1 and 2
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 the site-specific controlling document for HCVS out of service and compensatory measures.	The HCVS out of service and compensatory measures were included in a revision to OPLP-01.4, Fukushima FLEX System Availability, Action, and Surveillance Requirements. The 0PLP-01.4 revision was issued concurrently with Revision 3 to the Severe Accident Guidelines during the spring 2017 Unit 2 refueling outage. This procedure will be revised to incorporate Unit 1 HCVS requirements when that unit's HCVS modifications are installed in accordance with the milestone schedule reported in the BSEP Overall Integrated Plan. The 0PLP-01.4 procedure revision that incorporates Unit 2 HCVS is available for review on the ePortal.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The guidelines and procedures for HCVS out-of-service and compensatory measures are complete for Unit 2 and consistent with the guidance in NEI 13-02. Unit 1 procedures will be revised following the installation of the HCVS modifications and will follow the same guidance as Unit 2, consistent with the guidance in NEI 13-02. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.13]
Phase 1 ISE OI 2 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 (i.e., unless a lower value is justified), and that the suppression pool and the	0FLEX-0035, Flow Capacity of BNP Hardened Wetwell Vent Units 1 & 2 at 1% Rated Power, provides the calculation showing that both units' hardened vents' flow capacity is greater than 1 % thermal power at design pressure which is lower than the primary containment pressure limit. This is documented in the results paragraph 4.4 on page 9 of 9 of the calculation. This calculation assumes that the new discharge check valve has a Cv	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation 0FLEX-0035, "Flow Capacity of BNP Hardened Wetwell Vent Units 1 & 2 at 1% Rated Power," Revision 0 assumed a rated thermal power of 2,923 MWt. The calculation	Closed [Staff evaluation to be included in SE Section 3.1.2.1]

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.	of at least 673. The full open Cv of the check valve is approximately 4000. Therefore, the vent pipe will pass at least 1 % thermal power equivalent. BNP-MECH-FLEX-002, Brunswick Nuclear Plant Containment Analysis of FLEX Strategies, is a MAAP [Modular Accident Analysis Program] calculation of the BSEP response to an extended loss of AC power (ELAP) event initiated from full power. The MAAP results also show that containment pressure is rapidly reduced and is maintained below design pressure and primary containment pressure limit (PCPL). This is best seen in the graph on page 4 of Appendix 7 (pdf page 55) which is a plot of Run 1 containment response. Run 1 models the BSEP procedural guidance for the FLEX event Support documents are available for review on the ePortal.	used the saturated vapor enthalpy at containment design pressure of 62 per sqaure in gauge (psig). The required vent capacity is 84,420 lb/hr. The calculation conservatively used a flow of 85,000 lb/hr. The calculation used AFT Arrow, version 4.0 computer program to model the vent system. The program determined the minimum Cv (flow coefficient) for the check valve is 673. The full open Cv of the check valve is approximately 4000. Therefore, the licensee's HCVS design will meet the 1% of rated thermal power requirement. No follow-up questions.	
Phase 1 ISE OI 3 Make available for NRC staff audit confirmation of the time it takes the suppression pool to reach the heat capacity temperature limit during ELAP with RCIC in operation.	BNP-MECH-FLEX-0002, provides the suppression pool (SP) response to the ELAP with operator actions. Initially, in this analysis, reactor core isolation cooling (RCIC) is aligned to the suppression pool (SP). In this analysis, after 1 hour, the SP is approaching the heat capacity temperature limit (HCTL), although it has not yet reached it. At this point, 1 hour, the operators begin a controlled cooldown to 450 psig using one safety relief valve (SRV). This reduces primary pressure while heating up the SP, but the net result is that the SP stays below the HCTL. At 2 hours, the operators further depressurize the reactor pressure vessel	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation BNP-MECH-FLEX-002, "Brunswick Nuclear Plant Containment Analysis of FLEX Strategies" determined the time for the suppression pool to reach HCTL during an ELAP with RCIC operating to be approximately 3.2 hours. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.1]

	(RPV) to 150-300 psig, which initially maintains the SP below HCTL. The exact time of reaching HCTL depends on the timing of SP heatup and the cycling of RPV pressure between 150 and 300 psig since the actual limit is a function of RPV pressure. Since pressure is cycled between 150 psig and 300 psig after hour 2, it is conservative to determine the time at which the SP temperature and level reach the HCTL at 300 psig using the 0EOP-01-NL, HCTL curve. During this time, SP level is slowly increasing as shown in BNP-MECH-FLEX-0002, but is about -2.4 feet. This puts the HCTL temperature at about 193°F in the SP, which is reached at about 3.2 hours. Support documents are available for review on the ePortal.		
Phase 1 ISE OI 4 Make available for NRC staff audit a description of the final ROS location.	The location for the remote operating station (ROS) is in the southeast corner of the Reactor Building (RB) 50'-0" elevation for Unit 1, and the northeast corner of the RB 50'-0" elevation for Unit 2. The ROS locations inside the RB are in a corridor just inside a door to the outside of the RB that will be blocked open in an ELAP. This door access provides a direct path to the Main Control Room (MCR) via the Radwaste Building roof. The evaluation of the ROS for temperature and radiation concerns is contained in the response to ISE open item #10.	The NRC staff reviewed the information provided in the 6-month updates. The ROS is in a location that is readily accessible and appears to support operation of the HCVS. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.4]
Phase 1 ISE OI 5	The primary operating station for the HCVS is and remains in the Main Control	The NRC staff reviewed the information provided in the 6-	Closed

Make available for NRC staff audit documentation that demonstrates adequate communication between the remote HCVS operation locations and the HCVS decision makers during ELAP and severe accident conditions.	Room (MCR) with the implementation of order EA-13-109. For each unit, the alternate operating station (also called the Remote Operating Station or ROS) is located just inside the Reactor Building (RB) at the 50-foot elevation, adjacent to a door to the outside. The door is in the southeast section of the RB for Unit 1 and in the northeast section for Unit 2. The MCR will direct operators to this alternate control location if required due to an inability to operate the HCVS valves from the MCR. In addition, operators will be dispatched to the backup pneumatic connections on each unit in order to connect the backup air compressor before the 24-hour nitrogen supply is exhausted. These HCVS activities may require communication with the MCR. As part of the response to NRC Order EA-12-049, BSEP assumed that permanently installed plant communications systems would not be available during an extended loss of AC power (ELAP). Instead, BSEP primarily utilizes an 800 MHz [Mega Hertz] radio system consisting of 500 hand-held radios for onsite communications. These radios are stored in reasonably protected buildings, including the FLEX Storage Building, to meet the requirements of EA-12-049. This information was provided in response to NTTF Recommendation 9.3, by a letter dated October 31, 2012 (ADAMS Accession No. ML12311A299) and supplemented by a letter dated February 22, 2013, Carolina Power & Light Company's and Florida Power Corporation's Response to Follow-Up	month updates and on the ePortal. The communication methods are the same as accepted in Order EA-12-049. No follow-up questions.	[Staff evaluation to be included in SE Section 3.1.1.1]

	Letter on Technical Issues for Resolution Regarding Licensee Communication Submittals Associated with Near-Term Task Force Recommendation 9.3 (ADAMS Accession No. ML13058A045). This information was assessed by the NRC staff and a Staff Evaluation was issued for this assessment. This was provided in Brunswick Steam Electric Plant, Units 1 and 2 - Staff Assessment in Response to Information Request Pursuant to 10 CFR 50.54(f)-9.3, Communication Assessment, dated April 4, 2013 (ADAMS Accession No. ML13093A341).		
Phase 1 ISE OI 6 Provide a description of the final design of the HCVS to address hydrogen Complete detonation and deflagration.	HCVS-WP-03, Hydrogen/Carbon Monoxide Control Measures (ADAMS Accession No. ML14302A066}, on page 2, lists the information that licensees shall provide with respect to strategies and options that "ensure the flammability limits of gases passes through the system are not reached." From HCVS-WP-03, page 2: 1. Declare option or options selected (valid for use of Options 3, 4 and/or 5) 2. List any deviations relative to the selected option(s) along with justification 3. Synopsis of venting operation and design 4. Sketch of vent path from associated PCIVs to release point, with delineation of which option applies to each portion of the vent system The information is provided below and was included in the December 2015 six-month update to the Overall Integrated	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee's design is consistent with Option 5 of the endorsed white paper HCVS-WP-03. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.11]

	Plan (ADAMS Accession No. ML16020A064). 1. BSEP has chosen option 5 which is to install a downstream check valve to prevent air from being drawn into the vent pipe when venting is stopped. 2. BSEP is not planning any deviations relative to option 5. 3. BSEP procedures contain guidance to open the hardened vent if plant conditions require it to prevent containment pressure exceeding the Primary Containment Pressure Limit. The vent will remain open until alternate reliable containment heat removal is established unless there is some condition or event that would require it be closed. There are no procedure steps that direct the vent be cycled to maintain a certain containment pressure band. The vent design is described in the BSEP OIP. 4. The sketch of the vent path with delineation of which option applies is available for review on the ePortal. Piping downstream of the second containment isolation valve, CAC-V216, is protected by the check valve (Option 5). The final HCVS design installs a check valve in the piping slightly below the Reactor Building roof as discussed in item 5 of the table on page 12 of HCVS-WP-03. The check valve will be mounted near the roof to minimize seismic effects, and will be less than 30 pipe diameters from the end as discussed in HCVS-WP-03 page 35. The BSEP check valve will minimize leakage of air into the HCVS piping such that a flammable mixture will		

	not occur while venting has been stopped without alternate containment heat removal. Just downstream of the check valve, BSEP will install a low pressure, 13 psig, rupture disk that will allow check valve testing, but will not prevent containment venting to avoid the primary containment pressure limit (PCPL). As part of the modifications, the new check valve will have test ports above and below it that will allow testing to verify that the valve opens and allow testing to verify that the valve leaks less than an amount that would allow a combustible mixture to occur in the pipe. Support documents, including a sketch of the vent path with delineation of which option applies, are available for review on the ePortal.		
Phase 1 ISE OI 7 Make available for NRC staff audit seismic and tornado missile final design criteria for the HCVS stack.	BSEP evaluated the HCVS stack for all Beyond-Design-Basis-External Events in Engineering Change (EC) 299559, Evaluation of the Hardened Wetwell Vent for Beyond-Design-Basis External Events, attachment Z01RO. This evaluation is available for review on the ePortal. This evaluation was provided as part of the BSEP FLEX audit in 2014, and was accepted for the tornado missile hazard disposition in Section 3.4 of Brunswick Steam Electric Plant, Units 1 and 2 - Report for the Audit Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051, March 31, 2015 (ADAMS Accession No. ML15082A155).	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Engineering Change (EC) 299559, "Evaluation of the Hardened Wetwell Vent for Beyond Design-Basis External Events," Revision 0, evaluated the HCVS stack. The hardened wetwell vent (HWV) is routed through a seismic isolation space between the Reactor Building and the Turbine Building. The HWV is protected from tornado missiles by the Reactor Building and the Turbine Building. The HWV reenters the Reactor Building and	Closed [Staff evaluation to be included in SE Section 3.2.2]

	The HCVS stack is part of the Hardened Wetwell Vent system that is evaluated in EC 299559. Sections 3.1.1 and 3.1.2 state the seismic design input and hazard criteria. Section 3.2.1 dispositions the seismic hazard. Section 3.1.5 states the design criteria for the tornado missile hazard (along with the high wind hazard). Section 3.2.4 dispositions the tornado missile hazard (along with the high wind hazard).	exits through the roof. The Reactor Building provides tornado missile protection. The licensee looked at potential wind-driven missiles and concluded that none of the potential missiles could be accelerated to speeds sufficient to damage the HWV. No follow up questions.	
Phase 1 ISE OI 8 Make available for NRC staff audit documentation of the HCVS nitrogen pneumatic system design including sizing and location.	Calculation 0RNA-0001, "Instrument Air Nitrogen Backup System Volume Requirements," provides the backup nitrogen system usage calculation and adequacy verification. On pages 5 and 6, the base calculation determines usage by the Safety-Relief Valves (SRVs), the Reactor Building to Suppression Chamber vacuum breaker valves, the Hardened Wetwell Vent Valves, and leakage. The total usage is determined to be 910 standard cubic feet against an available volume of 961 cubic feet (page 4). However, this usage was over a 22-hour period, vice the 24-hour period required by EA-13-109. As part of the BSEP response to EA-13-109, 2 bottles were added to each unit's Backup Nitrogen System, on each of 2 divisions. Appendix A of this calculation was created to demonstrate that the system has 24 hours worth of capacity. Appendix A shows that, with the additional bottles being added, there is enough nitrogen in Division 2 alone to supply 24 hours of nitrogen including leakage assumptions, HCVS valve	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation 0RNA-0001 discusses the pneumatic design and sizing. This calculation determined the required number of nitrogen cylinders needed in the backup nitrogen system for vent 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.	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

	cycling, SRV cycling, and containment vacuum breaker cycling. The safety-related Backup Nitrogen System bottles are located in seismically qualified racks (sections B.5.5 and B.5.10 of ECs 290410, Hardened Containment Vent System - Backup Nitrogen Bottles Unit 2, and 292338, Hardened Containment Vent System - Backup Nitrogen Bottles Unit 1) on the 50' elevation of the Reactor Building. The locations are shown on drawing F-02503 for Unit 2 and F-25003 for Unit 1. These bottles are always lined up to supply the HCVS vent valves if required so that no operator actions are required at the bottle racks. If the HCVS valves cannot be operated electrically, the operators can open them from the Remote Operating Station, located as shown on drawing F-02503, without approaching primary containment or the vent valves themselves (which are approximately 60 feet below the ROS, and in the area of the vent pipe, across the Reactor Building from the ROS). For pneumatic makeup after the backup bottles are depleted (later than 24 hours), the FLEX air compressor will be connected to the Backup Nitrogen System.		
Phase 1 ISE OI 9 Make available for NRC staff audit documentation of HCVS incorporation into the FLEX diesel generator loading calculation.	As described in 31116-CALC-E-001, "FLEX Diesel Generator Sizing Calculation," the bounding expected load for the FLEX diesel generators (DGs) is 367.4 kW. Taking a 25% margin, the required maximum output of the FLEX DG must be at least 460 kW. A nominal 500	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The licensee stated that all electrical power required for	Closed [Staff evaluation to be included in SE Section 3.1.2.6]

	kW FLEX DG meets this requirement. As discussed in this calculation, the major load is the battery chargers, and they have completed re-charging the batteries within 11 hours. The battery chargers represent 288 KW of the 367.4 kW maximum load. The majority of the loads initially aligned to the FLEX DGs are battery chargers and UPS, as described in Calculation 31116-CALC-E-001. The FLEX DGs are oversized for the load profile which helps minimize any effects from non-linear loading. This can be seen since the diesel generators are rated 500kW, but the maximum draw, for FLEX critical loads, including the non-linear loading will be less than 380kW. The non-linear loading from the battery chargers quickly drops off after batteries are fully recharged. This can be seen from the load profiles in Calculation 31116-CALC-E-001 where the power draw to the chargers drops below 20% of rated load after 6 hours. While the exact loading of the HCVS has not been incorporated into the FLEX DG loading calculations above, inspection of the HCVS power supply demonstrates that the HCVS load is insignificant to the FLEX DGs given the amount of load margin available. Calculation BNP-E-6.076-ICC-001, “Hardened Containment Vent System - Unit 2 Power Distribution,” adds the HCVS Radiation Monitor to the loading of the associated battery, 1.124 amps at 24 VDC [volts direct current] as shown on page 3 of Attachment 1 of BNP-	operation of HCVS components is provided by the Division 2 24/48 VDC battery and battery chargers. The battery sizing calculation BNP-E-6.125 confirmed that the 24/48 VDC battery has a minimum capacity capable of providing power for 24 hours without recharging, and therefore is adequate. The licensee provided 31116-CALC-E-001, which discusses re-powering of the 24/48 battery chargers using the FLEX DG. No follow-up questions.	

	E-6.076-ICC-001. Calculation BNP-E-6.125, "24/48 VDC Battery Allowable Discharge Rate for HCVS during an ELAP," contains the additional loading of the three instrument loops that will be powered by the HCVS distribution. These three instrument loops total 0.06 amps at 24 VDC as shown on page 1-1 of BNP-E-6.125. Therefore, the total load of the HCVS distribution is approximately 1.184 amps at 24 VDC or a little more than 28 watts. The 28 watts is insignificant to the FLEX DG load since the margin available in the FLEX DGs, even when the safety-related battery chargers are in service is approximately 132.6 KW. The full one-hour load on the Division 2 24/48 VDC batteries is approximately 20 amps per BNP-E-6.076-ICC-0001. This represents a load of 20 Ax 24 VDC = 480 watts. Assuming the FLEX DGs are required to carry the full load of the Division 2 24/48 VDC batteries through the charger, the additional 480 watts is also insignificant to the 132.6 kW of available capacity. Therefore, the FLEX DGs are fully capable of carrying the HCVS loads at any time they are energized.		
Phase 1 ISE OI 10 Make available for NRC staff audit an evaluation of temperature and radiological conditions to ensure that operating personnel can safely access and operate control and support equipment.	Operator actions for HCVS may be required at the following operating locations during an ELAP (see "Operator Action Maps.pdf" available for review on the ePortal): 1. Main Control Room (MCR) (primary operating location) 2. Control Building 49' elevation (location of HCVS power supply transfer switches)	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation RWA-L-1312-003 contains the Control Building GOTHIC room heatup analysis for the ELAP event. This calculation assumes selected doors open at	Open [Staff evaluation to be included in SE Sections 3.1.1.2 and 3.1.1.3]

	3. Outside of the RB, FLEX instrument air supply and refueling of FLEX compressor for long-term pneumatic supply 4. Outside of the RB, FLEX Diesel Generator (DG) enclosure to refuel the FLEX DGs for long-term electrical supply 5. Reactor Building (RB) - 50' elevation at the Remote Operating Station (ROS) Main Control Room and Control Building 49' - Temperature Evaluation Calculation RWA-L-1312-003, BNP Control Building (CB) FLEX Room Heat-up Analysis, contains a Control Building GOTHIC room heatup analysis for the ELAP event. This analysis takes no credit for operator action for the first six hours (other than opening panel doors) at which time the outside doors to the Control Building are opened and fans are started to force outside air through the building. This is a FLEX action evaluated as acceptable in response to NRC Order EA-12-049. This action is represented by Case 4 as shown in Table 7 on page 18 of 51. The results are tabulated on page 19 of 51 in Table 8. The results show ambient temperatures being maintained below 124°F for all spaces in which there may be operator actions for HCVS. Per HCVS-FAQ-06 in NEI 13-02 Appendix J, FLEX strategies that are not specific to HCVS can be credited as previously evaluated for FLEX. This temperature is judged acceptable for the simple and non-physical operator actions (i.e., switch manipulation, meter reading) required for HCVS operation during an ELAP event.	the start of the event. The Control Room temperature peaks at 116°F. After compensatory actions of opening select doors and installing a portable fan, the temperature drops below 95°F. For the remainder of the 72 hours modeled, the Control Room temperature cycles between roughly 95°F and 105°F following outdoor air diurnal temperature variation. Calculation BNP-MECH-FLEX-0001 contains the Reactor Building GOTHIC room heatup analysis for the ELAP event. This calculation determines the maximum temperature at the ROS to be 121°F. The operator actions will take place near the 50' door near the ROS so the local temperature at this location will be close to ambient outside the RB. Stay time in ROS will be limited. Procedures identify requirements for hot area work. Ice vests will be available as need. The MCR and its boundary are acceptable for radiological conditions without further evaluation for HCVS actions per NEI 13-02, Rev.1, HCVS-FAQ-01. For the ROS, the licensee provided a qualitative argument as to why the dose is not a concern. During the audit call on	

	Main Control Room and Control Building 49' - Radiation Evaluation The MCR and CB 49' (49' is adjacent to the MCR and inside the MCR boundary) are acceptable for radiological conditions without further evaluation for HCVS actions per NEI 13-02, Rev.1, HCVS-FAQ-01. Outside Areas for Pneumatic Makeup, Electrical Supply, and Refueling Activities Pneumatic makeup location The pneumatic supply for the first 24 hours of the ELAP event comes from the safety-related Backup Nitrogen System. No operator actions are required to supply pneumatics in the first 24 hours. On both units, there is a makeup station for the backup nitrogen system in the seismic isolation space between the Reactor Building (RB) and Turbine Building (TB) (see Operator Action Maps.pdf available for review on the ePortal). Per the response to order EA-12-049, portable FLEX compressors will be moved to outside locations near these makeup stations. Since the locations are outside the RB, there is no possible effect from RB heatup due to the ELAP. The compressors can be safely operated and refueled from this outside location as they will be shielded from the vent pipe by at least two of the RB concrete walls (three feet thick each) and no actions are required in the RB to supply the long-term pneumatic supply.	February 22, 2018, the NRC staff requested that the licensee perform a dose calculation. This item will remain open until the licensee provides the NRC staff a dose calculation, which shows the integrated radiation dose due to HCVS operation should not inhibit operator actions needed to initiate and operate the HCVS during an ELAP with severe accident conditions.	

	The makeup connections in the seismic isolation spaces are near the vent pipes (more so on Unit 2 than Unit 1) and possibly subject to gamma dose from the pipe once venting starts. Therefore, the connections of hose to the makeup stations in the seismic isolation space will be made before venting starts at approximately 17. 7 hours. Electrical makeup location The HCVS electrical supply for the first 24 hours is from the station 24/48 VDC battery system. This backup power supply is aligned at the 49-foot elevation of the Control Building adjacent to the MCR. As previously stated, this location is in the Control Building inside the MCR boundary and is acceptable for the duration of the event. The long-term electrical supply for the HCVS is from the FLEX Diesel Generators which can repower the normal supply buses to the HCVS controls and instruments or re-power the 24/48 VDC battery chargers. The FLEX Diesel Generators are located in the FLEX DG enclosure which is east of the RBs and the Emergency Diesel Generator (EDG) building (see Operator Action Maps.pdf available for review on ePortal). The location is on the opposite side of the RBs from the HCVS pipes and outside the RBs so that there are no concerns with operation of the FLEX DGs including refueling operation. No electrical lineups needs to be made in the RB for the FLEX DG to supply the needed HCVS		

	components, only inside the EDG Building which is not a dose or temperature concern area. Remote Operating Station - Temperature Evaluation Calculation BNP-MECH-FLEX-0001 documents the Reactor Building Heatup Analysis under ELAP conditions in which all ventilation, heating and cooling are deenergized. This analysis was used for development of the FLEX actions per order EA-12-049, but since the same Extended Loss of AC Power (ELAP) conditions apply to the EA-13-109 order, this analysis can be used to estimate the temperature at the ROS for HCVS purposes. Even though EA-13-109 requires the consideration of a severe accident, the existence of core damage and possible vessel breach will have no effect on the temperature at the ROS. The applicable case in BNP-MECH-FLEX-0001 is case 1 which models the operator actions in an ELAP. The GOTHIC analysis results in Table 4 (page 23) show that the maximum temperature on the 50' elevation is 121°F. The actions at the ROS will be to open or close a maximum of three ½ inch valves so that they are expected to take less than 5 minutes. Furthermore, the operator will be entering the RB through the 50' door near the ROS so that the local temperature will be close to ambient outside the RB. These temperatures, coupled with the short duration of action, are judged acceptable.		

	Remote Operating Station - Radiation Evaluation The bottom of the active core region is at 51' elevation. Therefore, an operator would be roughly at core elevation while at the ROS. The shielding provided by the vessel, bio-shield, Primary Containment (PC), and distance from the core results in the 50' door location being a low-dose-rate-waiting area during normal full-power operation. The Primary Containment wall alone provides six feet of concrete shielding. Since the core is shutdown for the ELAP event, the dose rates from the core area will be lower than during operation. The existence of core damage with possible reactor pressure vessel breach will not raise the dose levels at the ROS. If the core were to melt through the lower vessel head, there would be loss of shielding from the vessel, however there would be additional distance to the ROS and additional concrete shielding provided by the pedestal. Any gap release to the suppression pool will contribute to RB dose rates, however the ROS is on the 50' elevation, two floors above the torus. Therefore, the dose rate at the ROS due to the torus will be insignificant due to the 5 feet of concrete below the ground floor as well as the additional concrete and distance afforded by the location being on the 50-foot elevation. Likewise, any gap release that migrates back to the Primary Containment, will be shielded from the ROS by the 6' thick Primary Containment		

	wall. In addition, the ROS is approximately 50 feet away from the PC wall. Support documents, including the Operator Action Maps.pdf, are available for review on the ePortal.		
Phase 1 ISE OI 11 Make available for NRC staff audit descriptions of all instrumentation and controls (i.e., existing and planned) necessary to implement this order including qualification methods.	A list of instruments and controls necessary to implement EA-13-109 with their descriptions and qualification methods is provided in the December 2016 Six-Month Status Report by letter dated December 15, 2016 (i.e., ADAMS Accession No. ML 16365A007).	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The existing plant instuments required for HCVS (i.e. wetwell level instruments and drywell pressure instruments) meet the requirements of Regulatory Guide (RG) 1.97. The licensee provided a list of HCVS instruments and controls (I&C) a brief description of each component, the component identification number, the component make and model number and the qualification method. The staff's review indicates that the I&C components are consistent with the guidance in NEI 13-02 and its qualifications meet the order requirements. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.8]
Phase 1 ISE OI 12 Clarify whether the seismic reliability demonstration of instruments, including valve position indication, vent pipe	Existing equipment installed prior to RG 1.97 is qualified in accordance with original licensing basis and IEEE 344-1971. Equipment installed after RG 1.97 is qualified to IEEE 344-1975. Therefore, the BSEP HCVS instruments will be	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 3.1.1.4]

temperature instrumentation, radiation monitoring, and support system monitoring will (be) via methods that predict performance described in [Institute of Electrical and Electronics Engineers] IEEE-344-2004 or provide justification for using a different revision of the standard.	qualified to IEEE 344-1971 or 1975. The exception is the new 24 VDC voltmeter being installed for EA-13-109 response is qualified to IEEE-344-2004 as this vendor only provides qualification to that version. See ISE Open Item #14 for more details on the instrument qualifications.	The licensee stated, in part, that the seismic qualification method was dependent on the time of original installation and that most I&C equipment is pre-existing except the voltmeter for the new HCVS battery. The staff confirmed the individual component seismic qualification methods in the I&C component list provided. No follow-up questions.	
Phase 1 ISE OI 13 Make available for NRC staff audit a justification for not monitoring HCVS system pressure as described in NEI 13-02.	While NEI 13-02 paragraph 4.2.4.5 provides an acceptable approach for HCVS monitoring that includes vent pipe pressure, BSEP has not included HCVS vent pipe pressure. If the HCVS is not in service, a vent pipe pressure indicator would not provide useful information. If the HCVS is placed in service, BSEP has several indicators that will reliably indicate the status of containment and of the HCVS. The following indicators are already qualified for post-accident conditions or are qualified per the requirements of EA-13-109. 1. Drywell pressure 2. HCVS valve position indication 3. HCVS pipe temperature 4. HCVS pipe radiation level 5. Suppression Pool level These five instruments provide sufficient information for the operators to monitor the status of the vent system without the addition of a vent pipe pressure indicator.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The NRC staff determined that HCVS valve position indication, HCVS pipe temperature, HCVS radiation level and drywell pressure are sufficient indication to determine the vent is operating as expected. Supression pool level is needed to determine the amount of freeboard before operating the vent. No follow-up questions.	Closed [Staff evaluation to be included in SE Section 3.1.2.8]
Phase 1 ISE OI 14	The list of components and their evaluations is available for review on the	The NRC staff reviewed the information provided in the 6-	Pending

Make available for NRC staff audit the descriptions of local conditions (i.e., temperature, radiation and humidity) anticipated during ELAP and severe accident for the components (e.g., valves, instrumentation, sensors, transmitters, indicators, electronics, control devices, etc.) required for HCVS venting including confirmation that the components are capable of performing their functions during ELAP and severe accident conditions.	ePortal in spreadsheet HCVS ISE OPEN ITEM 14.xlsx. The components in the table boxes with no background color are new for EA-13-109 compliance. The components with the blue background color are existing plant equipment that meet the current design basis of the plant. All components are evaluated for temperature, humidity integrated radiation, and seismic adequacy. The estimates of temperature in the Reactor Building at the various locations are based on GOTHIC analyses of the ELAP event for the Reactor Building. Reactor Building humidity is assumed to rise to 100% due to boiling from the spent fuel pool in less than 7 days. For the Control Building, the temperature estimates are based on a GOTHIC analysis that assumes zero humidity, thereby maximizing temperature response. The Control Building humidity is assumed to reach a maximum of 91% which is based on the historic maximum humidity of the ambient air (used to ventilate the Control Building in the FLEX strategies) from UFSAR Table 2-24. All components are either seismically qualified to IEEE-344-1975 (the battery voltmeter is new and is qualified to IEEE-344-2004) or have been evaluated as seismically rugged so that they will perform their function following a seismic event. The estimate of radiation dose at any component is based on the results presented in Table 2 of HCVS-WP-02 as scaled to BSEP plant specifics. For this evaluation of components, the 4-hour time	month updates and on the ePortal. The NRC staff reviewed the table provided on the eportal. The staff anticipates a similar table in the FIP (on the docket) to close the item. No follow-up questions.	[Staff evaluation to be included in SE Section 3.1.1.4]

	step was chosen for the pipe dose rates and the dose rate is held constant rather than accounting for decay. In the BSEP MAAP analysis 4 hours is before the vent would be opened to avoid PCPL. Since the dose rate decreases after the 4-hour time step, it is conservative to use this dose rate. For valves and other components that are in or on the pipe, the 1' dose is used and integrated over a 7-day period. For other components in the Reactor Building such as pressure transmitters, the 3' dose is used. This is conservative because these instruments are, in fact, not near the vent pipe and are shielded from the vent pipe by the 3' thick Reactor Building wall or the Primary Containment wall. The Primary Containment wall also shields these instruments from the airborne activity in the Primary Containment. As with the 1' dose components, this 3' dose is integrated over the 168-hour period with no allowance for decay. The resulting total integrated dose (TID) are then compared to the qualification total integrated dose for each susceptible component. All components are confirmed capable of performing their functions during ELAP and severe accident conditions. Support documents, including spreadsheet HCVS ISE OPEN ITEM 14.xlsx, are available for review on the ePortal.		
Phase 1 ISE OI 15 Make available for NRC staff audit documentation of an	BSEP procedure 0EOP-02-PCCP, Primary Containment Control Procedure, directs opening the hardened wetwell vent valves before reaching the primary	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal.	Closed

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.	containment pressure limit (PCPL) of 70 psig. Therefore, the maximum opening dip is 70 psid (containment to atmosphere). Calculation BNP-MECH-AOV-DP-CAC, in the table in Section 4.0, page 12 of this calculation, confirms that 70 psid is the maximum expected opening differential pressure. BNP-MECH-1-CAC-V7-AO and BNP-MECH-2-CAC-V7-AO contain the Air Operated Valve (AOV) calculations for the inboard wetwell purge valve on each unit. Section 4.1.1 contains a table of minimum margins for these valves. The minimum opening margin for 1-CAC-V7 is 12.7%, and for 2-CAC-V7 is 19.5%. BNP-MECH-1-CAC-V216-AO and BNP-MECH-2-CAC-V216-AO contain the AOV calculations for the hardened wetwell vent valve on each unit. Section 4.1.1 contains a table of minimum margins for these valves. The minimum opening margin for 1-CAC-V216 is 33.8%, and for 2-CAC-V7 is 25.7%.	The NRC staff reviewed BNP-MECH-AOV-DP-CAC, "Differential Pressure Calculations for ½ - CAC-V7-AO, -V8-A0, -V216-A0 Inboard Suppression Pool Purge Exhaust, Outboard Suppression Pool purge Exhaust, and Hardened Wetwell Vent Isolation Air-Operated Valves," Revision 0, which discusses the valve/actuator information for the PCIVs. The NRC staff verified the actuator can develop greater torque than PCIV's unseating torque. No follow-up questions.	[Staff evaluation to be included in SE Section 3.2.1]
Phase 1 ISE OI 16 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	As shown and described in the BSEP OIP, the HCVS pipe taps off a 20-inch torus purge pipe, is routed outside the Reactor Building (RB) into the seismic isolation space between the RB and Turbine Building (TB), is routed up the outside of the RB, re-enters the RB at the 120' elevation, then exits through the RB roof. There is no penetration into any other building. The only interface between HCVS and any other system is through valves CAC-VB and CAC-V172. These two valves	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The wetwell vent for each unit utilizes CAC system valves CAC-V7 and CAC-V216 for containment isolation. CAC system containment isolation valves CAC-V8 and CAC-V172 are the only functional boundary valves between the HCVS and the downstream SBGT system.	Closed [Staff evaluation to be included in SE Section 3.1.2.12]

	connect the purge system to the Standby Gas Treatment System (located inside the RB) and are primary containment isolation valves (PCIV). Since they are PCIVs they are tested for leakage per 10 CFR 50 Appendix J. This testing methodology has been endorsed as an acceptable testing means in HCVS-FAQ-05. Therefore, it is expected that the potential for hydrogen gas migration to the SBGT system, which could lead to leakage into the RB, is minimized. As shown in Sketch 1 of the BSEP OIP (ADAMS Accession No. ML14191A687), the HCVS pipe is routed to the RB roof without any further connections to the RB atmosphere, to any system other than SBGT, or to any other building. This portion of the HCVS shall be leak tested in accordance with NEI 13-02. Since the pipe does not enter any other station building, there is no possibility of hydrogen gas migration into any other building. The HCVS piping is constructed of seamless type 304 stainless steel piping. The piping joints are a combination of welded and flanged connections. The piping was designed, fabricated and installed in accordance with ANSI B31.1 and is tested per NEI 13-02. Therefore, the HCVS piping can be considered leak tight and there is minimal potential for hydrogen to leak into the Reactor Building. The BSEP HCVS pipe is a connection off the wetwell purge line. The other branch	These valves are tested, and will continue to be tested, for leakage under 10 CFR 50 Appendix J as part of the containment boundary in accordance with HCVS-FAQ-05. The NRC staff's review of the proposed system indicates that the licensee's design appears to maintain hydrogen below flammability limits. No follow-up questions.	

	connections from this purge line contain automatic, fail-closed, containment isolation valves that are tested as part of 10CFR50, Appendix J, testing to ensure leakage is within limits (per HCVS-FAQ-05). The rest of the HCVS pipe is not connected to any other system and does not traverse any building other than the same unit Reactor Building. The HCVS pipe is sealed with flanges and closed valves, was pressure tested when initially installed and will additionally be tested after modifications for EA-13-109 compliance to ensure it is leak-tight.		
Phase 2 ISE OI 1 Licensee to confirm through analysis, the temperature and radiological conditions to ensure that operating personnel can safely access and operate controls and support equipment.	Operator actions may be required at the following operating locations during an ELAP (see "Operator Action Maps.pdf" located on the ePortal): 1. Main Control Room (MCR) (i.e., primary operating location) 2. Control Building 49' level (i.e., location of HCVS power supply transfer switches) 3. Outside of the Reactor Building (RB), FLEX instrument air supply and refueling of FLEX compressor for long-term pneumatic supply 4. Outside of the RB, at the FLEX Diesel Generator (DG) enclosure to refuel the FLEX DGs for long-term electrical supply 5. Reactor Building (RB) - 50' level at the Remote Operating Station (ROS) 6. East of the RB near the Condensate Storage Tanks (CST) to connect a hose for FLEX/Severe Accident Water Addition (SAWA) and stage the FLEX/SAWA pump. 7. At the outside wall of the RB (i.e., north for Unit 1 and south for Unit 2) at the	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. See comments for Phase 1 open item #10 for NRC staff's review considerations.	Open [Staff evaluation to be included in SE Sections 3.1.1.2 and 3.1.1.3]

	FLEX core bore for FLEX/SAWA pump discharge 8. Inside the RB at the 20' elevation (i.e., ground) Main Control Room and Control Building 49' -Temperature Evaluation: Vendor calculation RWA-L-1312-003, BNP CB FLEX Room Heat-up Analysis, contains a Control building GOTHIC room heatup analysis for the ELAP event (i.e., Reference EC 289577 Attachment Z52). This analysis takes no credit for operator action for the first six hours (i.e., other than opening panel doors) at which time the outside doors to the control building are opened and fans are started to force outside air through the building. This is a FLEX action evaluated as acceptable in response to NRC Order EA-12-049. This action is represented by Case 4 as shown in Table 7 on Page 18 of 51. The results are tabulated on Page 19 of 51 in Table B. The results show ambient temperatures being maintained below 120°F for all spaces except the electrical equipment rooms in which there are no actions for HCVS. Per HCVS-FAQ-06 in NEI 13-02 Appendix J, FLEX strategies that are not specific to HCVS can be credited as previously evaluated for FLEX. This temperature is judged acceptable for operator action during an ELAP event. Main Control Room, Control Building 49', and Control Building (Battery Rooms) - Radiation Evaluation:		

	The MCR and CB 49' (i.e., 49' is adjacent to the MCR and inside the MCR boundary) are acceptable for radiological conditions without further evaluation for HCVS actions per NEI 13-02, Rev.1, HCVS-FAQ-01. Areas for SAWA Injection, Pneumatic Makeup, Electrical Supply, and Refueling Activities SAWA Locations The source of water for SAWA is the Condensate Storage Tanks (CST), which were qualified for all external hazards for EA-12-049 response. Operators will connect a suction hose, stored with the pump in the FLEX Storage Building (FSB), to the tank and then to the pump. The pumps are staged outside and east of the RBs. From the pumps, hoses are run to the FLEX core bores on the outside of the RBs (north wall for Unit 1 and south wall of Unit 2). The Severe Accident Water Mitigation (SAWM) flow instrumentation is mounted on the pump itself. Since the pumps are outside the RB, there are no ambient temperature concerns for personnel action. Also since the pumps are outside the RB and on the opposite side of the RBs from the vent pipes, there is no concern for the radiation levels at the pumps from the vent pipe or damaged core which would still be inside the primary containment.		

	Likewise, the long term makeup for the CST is from the discharge canal weir. The pump staging location and hose runs for this makeup path are far from the RBs and the vent pipes, so there are no radiological concerns for CST makeup. Inside the RB, the SAWA pipe is aligned by opening three valves at the 20' (i.e., ground) level of the RB. These valves will be opened within the first hour after the start of the ELAP before there is any core damage or significant RB heatup. The use of the hard pipe eliminates the need for the operators to run hoses inside the RB during an event. In the case of a seismically-initiated event, the operators will also close one valve on the 80' level of the RB, so at most four valves need be operated to align the SAWA flow path inside the RB. Pneumatic makeup location The pneumatic supply for the first 24 hours of the ELAP event comes from the safety-related backup nitrogen system. No operator actions are required to supply pneumatics in the first 24 hours as the backup nitrogen system automatically aligns itself. On both units, there is a makeup station for the backup nitrogen system in the seismic isolation space between the RB and Turbine Building (TB) (i.e., see Operator Action Maps.pdf). Per the response to order EA-12-049, portable FLEX compressors will be moved to outside locations near these makeup stations. Since the locations are outside the RB, there is no ambient		

	temperature concern for personnel actions. The staging location is shielded from the vent pipe by a minimum of six feet of concrete, so that the vent pipe radiation is mitigated. Therefore, the compressors can be safely operated and refueled from this outside location. No actions are required in the RB to supply the long-term pneumatic supply. The makeup connections in the seismic isolation spaces are near the vent pipes (more so on Unit 2 than Unit 1) and possibly subject to gamma dose from the pipe once venting starts. Therefore, the hose connections to the makeup stations in the seismic isolation space will be made before venting starts at approximately 7-8 hours. The hose connections will be fitted with quick disconnects to aid in making this a simple action during event response. The hose connection timing will be validated per HCVS-FAQ-13. Electrical makeup location The HCVS electrical supply for the first 24 hours is from the station 24/48 VDC battery system. This backup power supply is aligned at the 49 foot level of the control building at the same panel as the HCVS Radiation Monitor, adjacent to the MCR. As previously stated, this location is in the control building inside the MCR boundary and is acceptable for the duration of the event. The long-term electrical supply for the HCVS is from the FLEX Diesel Generators which can re-power the normal supply buses to the		

	HCVS controls and instruments or re-power the 24/48 VDC battery chargers. The FLEX generators are located in the FLEX DG enclosure which is east of the RBs and the Emergency Diesel Generator (EDG) building (i.e., see Operator Action Maps.pdf). The location is on the opposite side of the RBs from the HCVS pipes and outside the RBs so that there are no concerns with operation of the FLEX DGs including refueling operation. No electrical lineups need be made in the RB for the FLEX DG to supply the needed HCVS components, only inside the EDG building which is not a dose or temperature concern area. Remote Operating Station - Temperature Evaluation: Calculation BNP-MECH-FLEX-0001 documents the Reactor Building Heatup Analysis under ELAP conditions in which all ventilation, heating and cooling are de-energized. This analysis was used for development of the FLEX actions per order EA-12-049, but since the same Extended Loss of AC Power (ELAP) conditions apply to the EA-13-109 order, this analysis can be used to estimate the temperature at the ROS for HCVS purposes. Even though EA-13-109 requires the consideration of a severe accident, the existence of core damage and possible vessel breach will have no effect on the temperature at the ROS. The applicable case in BNP-MECH-FLEX-0001 is Case 1 which models the operator actions in an ELAP. The		

	GOTHIC analysis results in Table 4 (i.e., Page 23 of 76) show that the maximum temperature on the 50' elevation is 121°F. The actions at the ROS will be to open or close a maximum of three ½ inch valves so that they are expected to take less than 5 minutes. Furthermore, the operator will be entering the RB from outside through the 50' airlock door near the ROS so that the local temperature will be close to ambient outside the RB. These temperatures, coupled with the short duration of action, are judged acceptable. Remote Operating Station - Radiation Evaluation: The bottom of the active core region is at 51' elevation. Therefore an operator would be roughly at core elevation while at the ROS. The shielding provided by the vessel, bio-shield, primary containment (PC), and distance from the core results in the 50' door location being a low dose rate area during normal full-power operation. The primary containment wall alone provides six feet of concrete shielding (i.e., drawing F-01132). Since the core is shutdown for the ELAP event, the dose rates from the core area will be lower than during operation. The existence of core damage with possible reactor pressure vessel breach will not raise the dose levels at the ROS. If the core were to melt through the lower vessel head, there would be loss of shielding from the vessel; however, there		

	would be additional distance to the ROS and additional concrete shielding provided by the pedestal. Any gap release to the suppression pool will contribute to RB dose rates, however the ROS is on the 50' level, two floors above the torus. Therefore, the dose rate at the ROS due to the torus will be insignificant due to the five feet of concrete below the ground floor (i.e., drawing F-01787) as well as the additional concrete and distance afforded by the location being on the 50' elevation. Likewise, any gap release that migrates back to the primary containment, will be shielded from the ROS by the six foot thick PC wall. In addition, the ROS is some 50 feet away from the PC wall.		
Phase 2 ISE OI 2 Licensee to provide the site-specific MAAP evaluation that establishes the initial SAWA flow rate.	BNP-MECH-FLEX-0005 documents the BSEP specific MAAP evaluation that verifies that an initial SAWA flow rate of 300 gpm [gallons per minute] is sufficient to protect containment as described in NEI 13-02, Rev. 1. Cases 1 and 2 are the cases that compare the containment response with a 300 gpm initial flow rate to the response with an initial flow rate of 500 gpm which is the maximum flow rate required by NEI 13-02. Section 7 starting on Page 25 of 178 presents the results of both analyses and compares them in Table 7-1 and Figures 7-1 to 7-5. As can be seen in these results, there is no significant difference between the two initial flow rates. The site-specific MAAP evaluation for SAWA, BNP-MECH-FLEX-0005, is available for review on the ePortal.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation BNP-MECH-FLEX-0005 is the BSEP specific MAAP evaluation that verifies that an initial SAWA flow rate of 300 gpm is sufficient to protect containment. The Electric Power Research Institute (EPRI) study (Technical Basis for Severe Accident Mitigating Strategies, 3002003301) assumes a 500 gpm SAWA injection flow. BNP evaluated a 300 gpm SAWA flow. The 300 gpm SAWA flow starts at 8 hours. At 12 hours SAWA flow is reduced to 100 gpm. The calculations assume a 2923 MWt. The calculation concludes that with the SAWA flow	Closed [Staff evaluation to be included in SE Section 4.1.1.3]

		indicated, wetwell venting is maintained, the peak suppression pool air space pressure is 84.7 per square inch absolute (psia) (70 psig) which is ≤ 70 psig acceptance pressure, and peak suppression pool water level is 17.1 feet. No follow-up questions.	
Phase 2 ISE OI 3 Licensee to demonstrate how instrumentation and equipment being used for SAWA and supporting equipment is capable to perform for the sustained operating period under the expected temperature and radiological conditions.	The instrumentation required for SAWA performance and monitoring consists of drywell pressure, torus level and the SAWA flow instrument. The only operating equipment for SAWA is the SAWA pump. The drywell pressure instruments are CAC-PT-1230 (i.e., one for each unit). The torus level instruments are CAC-LT-2601 (i.e., one for each unit). These instruments are safety-related and qualified to IEEE-323-1974 and IEEE-344-1975. They are qualified to 148.8°F but are expected to see no more than 132°F during the ELAP. They are qualified to 8.36E6 R total integrated dose and expected to receive no more than 9.13E5 R total integrated dose in the first seven days after the event starts. Therefore, both instruments are qualified for the conditions they would see during a severe accident. The drywell pressure and torus level instruments are powered from safety-related instrument buses that will be re-powered by the FLEX generators so that they will remain in service without offsite power or Emergency Diesel Generators.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The drywell pressure and torus level indications are RG 1.97 compliant and are acceptable as qualified. The SAWA flow instrument qualificaitons for temperature and radiation need to be included in Table 1 of the FIP. No follow-up questions.	Pending [Staff evaluation to be included in SE Sections 4.5.1.2 and 4.5.1.3]

	In the case that the FLEX generators are not initially available, both instruments can be repowered from the station's division two 24 VDC battery for at least 24 hours. The SAWA flow instrument is mounted on the FLEX/SAWA pump and is powered by the pump's generator. The SAWA pumps are stored in the FLEX Storage Building so that they will be available after the event. The SAWA pump is moved to the area outside of the RB, near the CST. This area is on the opposite side of the RB from the vent pipe so that radiation is not a concern. Additionally, since the pump is outside, it will not be in an area of excessive temperature due to the accident. The FLEX/SAWA pumps are refueled in accordance with BSEP's FLEX Support Guidelines developed in response to EA-12-049.		
Phaes 2 ISE OI 4 Licensee to demonstrate that containment failure as a result of overpressure can be prevented without a drywell vent during severe accident conditions.	BNP-MECH-FLEX-0005 documents the MAAP calculation of containment response over a 7 day period during a severe accident with only Severe Accident Water Addition and the wetwell vent in service for containment protection. This analysis demonstrates that the containment pressure remains below the Primary Containment Pressure Limit (PCPL) and the temperatures remain low enough that failure due to high temperatures is avoided. The base case is Case 1 in this report. Table 7-1 on Page 26 of 178 contains a table of results. The peak drywell air	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. Calculation BNP-MECH-FLEX-0005, "Brunswick Nuclear Plant MAAP 5.02 Analysis to Support SAWA Strategy," Revision 1 shows the MAAP calculation of containment response over a 7 day period during a severe accident with only SAWA and the wetwell in service for containment protection. The EPRI study (Technical Basis for Severe	Closed [Staff evaluation to be included in SE Section 4.2]

	space pressure is shown as 89.9 psia. This is the pressure at the time of vent opening. The MAAP program was set to initiate the vent based on wetwell pressure at PCPL of 84.7 psia. The drywell pressure is slightly higher based on the differential pressure across the wetwell water. However, it is clear from the graph of drywell pressure (PRB(2)) on Page 66 of 178 that the pressure drops rapidly when the vent is opened and never returns to a pressure near the PCPL. The peak drywell temperature is shown as 581°F on Table 7-1 on Page 26 of 178. As seen in the Figure 7-2 graph on Page 28 of 178, the drywell temperature (TGRB(2)) rapidly decreases when the vent is opened and never increases for the rest of the 7 day analysis period. The site-specific MAAP evaluation for SAWA, BNP-MECH-FLEX-0005, is available for review on the ePortal.	Accident Mitigating Strategies, 3002003301) assumes a 500 gpm SAWA injection flow. BNP evaluated a 300 gpm SAWA flow. 300 gpm SAWA flow starts at 8 hours. At 12 hours SAWA flow is reduced to 100 gpm. The calculation assumed a 2923 MWt. The calculation concludes that drywell temperature peaks at roughly 581°F and drops below 300° after SAWA flow is initiated. With 300 gpm SAWA followed by 100 gpm SAWA flow starting at 12 hours and continuing for 168 hours will not result in a large increase in the suppression pool water lever that could potentially challenge the operation of the HWV. No follow-up questions.	
Phase 2 ISE OI 5 Licensee to demonstrate that there is adequate communication between the MCR and the operator at the FLEX pump during severe accident conditions.	During a severe accident in which the SAWA/FLEX pump is in use to add water to containment, the pump will be located between the CST and the Reactor Building (RB). The primary accident communication means will be the 800 MHz radio system. As part of the response to NRC Order EA-12-049, BSEP assumed that permanently installed plant communications systems would not be available during an ELAP. Instead, BSEP primarily utilizes an 800 MHz radio system consisting of 500 handheld radios for onsite	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]

	communications. These radios are stored in reasonably protected buildings, including the FLEX Storage Building, to meet the requirements of EA-12-049. This information was provided in response to NTTF Recommendation 9.3, by a letter dated October 31, 2012 (ADAMS Accession No. ML12311A299) and supplemented by a letter dated February 22, 2013, Carolina Power & Light Company's and Florida Power Corporation's Response to Follow-Up Letter on Technical Issues for Resolution Regarding Licensee Communication Submittals Associated With Near-Term Task Force Recommendation 9.3 (ML1305BA045). This information was assessed by the NRC staff and a Staff Evaluation was issued for this assessment. This was provided in ML13093A341, Brunswick Steam Electric Plant, Units 1 and 2 - Staff Assessment in Response to Information Request Pursuant to 1 O CFR 50.54(f) - 9.3, Communication Assessment. The radios will enable the MCR to communicate with operators in the field at the SAWA/FLEX pump locations.		
Phae 2 ISE OI 6 Licensee to demonstrate the SAWM flow instrumentation qualification for the expected environmental conditions.	The SAWM flow instrumentation (i.e., sensor and meter) will be permanently installed on each SAWA/FLEX pump. Electrical power is provided by the pump's 12 VDC electrical system. The flow instrumentation is qualified for use on fire pumps in outside ambient conditions. SAWA pumps are staged outside for use, between the CST and Reactor Building.	The NRC staff reviewed the information provided in the 6-month updates and on the ePortal. The SAWA flow instrument qualificaitons for temperature and radiation need to be included in Table 1 of the FIP. No follow-up questions.	Pending [Staff evaluation to be included in SE Section 4.4.1.3]

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - REPORT FOR THE AUDIT OF LICENSEE RESPONSES TO INTERIM STAFF EVALUATIONS OPEN ITEMS RELATED TO NRC ORDER EA-13-109 TO MODIFY LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS DATED March 22, 2018

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