UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

February 10, 2016

EA-14-088

Jeremy Browning, Site Vice President
Entergy Operations, Inc.
Arkansas Nuclear One
1448 SR 333
Russellville, AR 72802-0967

SUBJECT: ARKANSAS NUCLEAR ONE - REVISED NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000313/2014010 AND 05000368/2014010

Dear Mr. Browning:

On February 23, 2015, Entergy Operations, Inc. (Entergy), the licensee for Arkansas Nuclear One (ANO), provided a response to the U.S. Nuclear Regulatory Commission's (NRC's) Final Significance Determination of Yellow Finding and Notice of Violation report (ML15023A076) issued on January 22, 2015. The response letter is docketed under ML15054A607. The documents can be found in the NRC's Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), which is accessible from the NRC's Web site at http://www.nrc.gov/reading-rm/adams.html. Specifically, the response letter stated that the licensee agrees that a performance deficiency existed and concurs with both violations, with the exception of one example in the Notice of Violation (Notice) involving the classification of the Unit 1 decay heat vault drain valves. The NRC reviewed the basis for the exception to the example of a violation using Part I, Section 2.3.7, of the NRC Enforcement Manual.

Separately, Dale James, ANO Recovery Director, provided comments regarding details described in the final significance determination in NRC Inspection Report 05000313/2014010 and 05000368/2014010 in an email dated February 11, 2015 (ML15079A381). On March 27, 2015, the NRC acknowledged the licensee's letter and email from your staff (ML15086A289). The NRC response to your staff's additional comments is contained in Enclosure 3.

Entergy Position on Classification of Drain Valves

Unit 1 decay heat vault drain valves ABS-13 and ABS-14 are manual valves that the licensee classified as non-safety related with no safety-related function. They are maintained closed and are verified closed prior to initiating post-accident reactor building sump recirculation to limit the spread of contaminated liquid outside the decay heat vaults. These valves isolate the vaults from the auxiliary building general area, and are opened as needed to drain water from the decay heat vaults to the auxiliary building sump.

The licensee classified these valves with a preventative maintenance classification of Non-Critical - Essential, this ensures that the licensee's engineering staff evaluates and concurs with any proposed changes. Preventative maintenance activities for valves ABS-13 and

ABS-14 include a periodic flush and leak rate test, which provides high confidence that the valves can prevent the backflow of water through the drain system. In addition, these valves are included in the licensee's augmented inspection program to manage aging effects.

Unit 1 was designed to meet the intent of the original General Design Criteria, which provided the basis for equipment classification. A significant portion of the Unit 1 construction was completed before additional classification guidance appeared in 1972 and 1973. The classification of structures, systems and components was documented in the Final Safety Analysis Report as the Q-List. The Q-List classified systems and major components rather than each individual component. At the time, the Atomic Energy Commission accepted this approach to the classification of Unit 1 equipment. The Safety Evaluation Report for the operating license restated the Unit 1 Final Safety Analysis Report definition of Class I. The Safety Evaluation Report concluded that "this method of classification meets our [NRC] requirements for the seismic and quality classification of safety-related structures, components and systems."

In response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS [Anticipated Transient Without Scram] Events," the licensee implemented a Component Level Q-list which detailed classification at the component level. During this timeframe, Safety Related (Q) was defined as those systems, structures or components that are relied upon to remain functional during or following design basis events to ensure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100, "Reactor Site Criteria." This definition is consistent with the definition of safety related identified in Generic Letter 83-28 and 10 CFR 50.2. The Unit 1 decay heat vault drain valves ABS-13 and ABS-14 were reviewed against this criterion and were classified as non-safety related by the licensee.

NRC Response

The NRC staff performed a detailed review of the licensee's comments concerning the Unit 1 decay heat vault drain valves documented in Violation 05000313/2014010-01 and 05000368/2014010-01. The violation stated that the licensee failed to design, implement, and maintain the features needed to implement the approved flood mitigation protection for the units.

The Unit 1 decay heat vaults contain the low pressure injection pumps, containment spray pumps, shutdown cooling heat exchangers, and associated piping which are part of the engineering safety features (ESF). Because these components perform functions needed to ensure the integrity of the reactor coolant pressure boundary and the capability to shut down the reactor and maintain it in a safe shutdown condition, these ESF components are safety-related and are required to be protected from the effects of flooding. The NRC-approved design specified that the decay heat vault boundaries would prevent water from outside the vaults from entering. However, during the events of March 31, 2013, water from a broken fire main flooded the lower level of the auxiliary building in Unit 1 and entered the Unit 1 train B decay heat vault from a leak through valve ABS-13.

The NRC staff previously determined that citing Appendix B to 10 CFR Part 50 is acceptable for non-safety-related components whose failure would impact the safety-related function of structures, systems, and components during design basis accidents. This position is most recently documented in a letter from Scott Morris, Director, Division of Inspection and Regional Support, to a member of the public, Mr. Brady, dated August 29, 2014 (Enclosure 2, ML14175A887). It states:

> The requirements of Appendix B to 10 CFR Part 50 apply “to all activities affecting the safety-related functions” of structures, systems, and components (SSCs). The NRC does not treat all SSCs designed to mitigate flooding at a nuclear power plant as safety related. However, if a flood mitigation SSC is designed to protect a safety-related SSC’s safety-related function during a design-basis flood, and the flood mitigation SSC would not have provided the required flood protection such that the safety-related function of the safety-related SSC would be affected during a design basis flood, then Appendix B would be applicable.

The staff concluded that, for its flood protection function, the Unit 1 decay heat vault drain valves are important to safety. The staff determined that, in addition to providing flood protection for the ESF equipment, the decay heat vaults provided a radiological barrier function to mitigate the consequences of an accident. The NRC-approved design credited the decay heat vaults as sealed radiological barriers. By specifying the design in this manner, the licensee did not include any radiological leakage from the components in the decay heat vaults in the accident dose calculations or control room habitability calculations. Also, the licensee did not monitor ESF components that could leak (e.g., pump seals, bolted flanges, relief valves, etc.) in an ESF leakage monitoring program to ensure that the combined leakage remains within the accepted offsite and control room dose calculations under accident conditions.

The NRC staff concluded that valves ABS-13 and ABS-14 mitigate the consequences of an accident that could result in offsite exposures and, therefore, meet the NRC’s criteria for those components to be classified as safety-related. The staff noted that the decay heat vaults’ ventilation dampers (supply dampers CV-7621 and CV-7622, and exhaust dampers CV-7637 and CV-7638) were appropriately classified as safety-related, but the access doors (Doors 5 and 6) were not appropriately classified as safety-related, as these components also had this accident function.

While the staff has concluded that Unit 1 valves ABS-13 and ABS-14 are required to be classified and treated as safety-related, because they provide an accident dose mitigation function, the violation in question specifically focused on the flood protection design of the plant. The staff has concluded that it would be appropriate to revise the Notice (violation A, example e) involving the drain valves. Enclosure 1 re-characterizes the Unit 1 decay heat vault drain valve violation example to focus on the failure to protect the safety-related function of the low pressure injection and containment spray systems during design flood events.

The failure to ensure the design requirements for the decay heat vault drain valves and access doors to support the radiological barrier function to limit the dose consequences to control room operators and the public was determined to be a separate violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” and to have very low significance because each component is subject to periodic testing. The final disposition of the radiological barrier function is addressed in NRC Inspection Report 05000313/2015004 and 05000368/2015004 (ML16028A146).

As mentioned in Enclosure 1, you are NOT required to respond to the revised Notice. If you choose to respond, clearly mark your response as a "Reply to a Notice of Violation; EA-14-088" and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, Region IV, 1600 East Lamar Boulevard, Arlington, Texas 76011-4511; and a copy to the NRC resident inspector at Arkansas Nuclear One within 30 days of the date of this letter.

In accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice and Procedures," a copy of this letter will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Marc Dapas,
Regional Administrator

Dockets: 50-313; 50-368
Licenses: DPR-51; NPF-6

Enclosures:
1 – Notice of Violation
2 – Response to Letter Regarding
Citing Flood Protection Violations
3 – Response to Licensee Staff's Comments

As mentioned in Enclosure 1, you are NOT required to respond to the revised Notice. If you choose to respond, clearly mark your response as a "Reply to a Notice of Violation; EA-14-088" and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, Region IV, 1600 East Lamar Boulevard, Arlington, Texas 76011-4511; and a copy to the NRC resident inspector at Arkansas Nuclear One within 30 days of the date of this letter.

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S:\RAS\ACES\ENFORCEMENT_EA CASES - OPEN\ANO_EA-14-088 exception to the NOV on decay heat vault drain valves\ACK_EA-14-088.docx ADAMS ACCESSION NUMBER: ML16041A383

× SUNSI Review By: JLDJ	ADAMS × Yes ☐ No	× Publicly Available ☐ Non-Publicly Available	× Non-Sensitive ☐ Sensitive

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Letter to Jeremy Browning from Marc Dapas dated February 10, 2016.

SUBJECT: ARKANSAS NUCLEAR ONE - REVISED NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000313/2014010 AND 05000368/2014010

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NOTICE OF VIOLATION

Entergy Operations, Inc.
Arkansas Nuclear One, Units 1 and 2

Dockets: 50-313, 50-368
Licenses: DPR-51, NPF-6
EA-14-088

During an NRC inspection conducted between February 10 and August 1, 2014, two violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the violations are listed below:

A. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions. Design changes shall be subject to design control measures commensurate with those applied to the original design.

Unit 1, Safety Analysis Report (SAR), Amendment 26, Section 5.1.6, "Flooding," defined the design basis and stated, in part, that seismic class 1 structures are designed for the maximum probable flood level at elevation 361 feet above Mean Sea Level (MSL). The Unit 1 SAR further stated that all seismic class 1 systems and equipment are either located on floors above elevation 361 feet or protected. Sections 5.3.2 and 5.3.5.2 of the SAR indicated that the auxiliary building and emergency diesel fuel storage vault, both quality-related, are seismic Class 1 structures.

Unit 2, Safety Analysis Report, Amendment 25, Section 3.4.4, "Flood Protection," defined the design basis and stated, in part, that seismic Category 1 structures were designed for the probable maximum flood. The Unit 2 SAR further stated that all Category 1 systems and equipment are either located on floors above elevation 369 feet, or protected. Table 3.2-2, "Seismic Categories of Systems, Components, and Structures," of the Unit 2 SAR indicated that the auxiliary building and emergency diesel fuel storage vault, both quality related, are seismic Class 1 structures.

Unit 1, Safety Analysis Report, Amendment 26, Section 5.3.2, "Auxiliary Building," stated, in part, that the floor area at elevation 317 feet containing engineered safeguards equipment, was partitioned into separate rooms to provide protection in the event of flooding due to a pipe rupture.

Contrary to the above, as of March 31, 2013, the licensee failed to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions and that design changes were subject to design control measures commensurate with those applied to the original design. Specifically, the licensee failed to assure that safety-related equipment below the design flood level was protected in the following examples:

a. The licensee failed to include a procedural step to install a blind flange in a ventilation duct that penetrated the Unit 1 auxiliary building below the design flood level.

b. The licensee failed to design the floor drain system with isolation capability so that the drain piping from the turbine building and radwaste storage building, which are non-flood protected structures, would not allow water to drain into the Unit 1 auxiliary building in the event of a flood.

c. The licensee failed to design the Unit 1 Hatch 522 and Unit 2 Door 253, which allow access to the area between the auxiliary buildings and containment buildings, to prevent water intrusion during a design basis flood event.

d. The licensee failed to seal open penetrations into the Unit 1 auxiliary building below the design flood level that were created when the licensee abandoned portions of the waste solidification system.

e. The licensee failed to assure that the Unit 1 decay heat vault drain valves loss of function during a design flood would not impact the safety-related functions of the low pressure injection and containment spray systems.

B. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Unit 1 Quality Drawing A-304, Sheet 1, "Wall and Floor Penetrations Key Plan," Revision 1, and Unit 2, Quality Drawings A-2002, "Architectural Schematic, Fire and Flood Protection Plans and Sections," Revision 10, prescribed walls, ceilings, and floors as flood barriers that required seals.

Unit 1, Quality Drawing A-337, "Wall and Floor Penetrations Enclosure Details," Revision 9, and Unit 2 Quality Drawing Series E-2073, "Electrical Penetration Sealing Details," Revision 3, prescribed conduit seal installation details that would act as a barrier to flood water. Unit 2 Quality Drawing Series A-2600, "Fire Barrier Penetration Seal Details," Revision 5, prescribed pipe penetration seal details that would act as a barrier to flood water.

Contrary to the above, as of March 31, 2013, the licensee did not accomplish activities affecting quality in accordance with documented instructions, procedures, or drawings. Specifically, the licensee failed to assure that safety-related equipment below the design flood level was protected in the following examples:

a. The licensee failed to install seals in conduits that penetrated flood barriers for the Unit 1 and Unit 2 auxiliary and emergency diesel fuel storage buildings.

b. The licensee failed to install seals in piping that penetrated flood barriers for the Unit 2 auxiliary building extension.

c. For the Unit 1 and Unit 2 auxiliary building hatches and building expansion joints between the building and containment, the licensee failed to provide appropriate seal inspection criteria, establish a replacement frequency for the seals, and develop post-maintenance test procedures to verify the effectiveness of the seals after they were reinstalled.

These violations are associated with a Yellow Significance Determination Process finding for Units 1 and 2.

The NRC has concluded that information regarding the reason for the violations, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in a letter from the licensee dated February 23, 2015, (ML15023A076). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation; EA-14-088" and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, Region IV, 1600 East Lamar Boulevard, Arlington, Texas 76011-4511; and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Revised this 10th day of February 2016

RESPONSE TO LETTER REGARDING CITING FLOOD PROTECTION VIOLATIONS

Enclosure 2

August 29, 2014

Mr. Joseph Brady
7726 Turnberry Lane
Stanley, NC 28164

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION RESPONSE TO LETTER REGARDING CITING FLOOD PROTECTION VIOLATIONS

Dear Mr. Brady:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your correspondence to Chairman Macfarlane dated October 26, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14100A257). You questioned the NRC staff's regulatory basis for citing flood protection violations against Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and whether these issues should have received an evaluation under 10 CFR 50.109, "Backfitting." You further provided specific examples to demonstrate your concerns.

We appreciate your perspectives. The staff recognizes the importance of flood protection, and we have increased our regulatory focus in this area following the 2011 Fukushima Dai-ichi accident in Japan and the 2011 Missouri River flooding at the Fort Calhoun Station. The NRC continues to look for opportunities to improve our programs to achieve the Principles of Good Regulation (Independence, Openness, Efficiency, Clarity, and Reliability) and the goals of the Reactor Oversight Process (Objective, Risk-informed, Predictable, and Understandable). Based on your concerns, we reviewed the basis for each citation for the specific examples you provided, the associated regulations, applicable Inspection Manual guidance, and associated generic communications.

The NRC's inspection and enforcement programs are designed to encourage licensees' prompt identification and comprehensive correction of violations of NRC requirements. The NRC recognizes that in some instances multiple applicable violations are associated with a given performance deficiency, and the inspectors and their management determine the most applicable requirement to cite in a notice of violation. Inspection reports and violations are documented in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," (ADAMS Accession No. ML12244A483) and each report is reviewed by regional management prior to issuance. The NRC uses IMC 0609, "Significance Determination Process [SDP]," (ADAMS Accession No. ML101400479) to determine the safety or security significance of the inspection finding (i.e., performance deficiency). For those violations associated with inspection findings that could be greater than very low safety significance, the NRC convenes a Significance and Enforcement Review Panel (SERP) to review the preliminary significance and basis. The SERP decision-makers comprise management from the applicable regional office, the Office of Nuclear Reactor Regulation, the Office of Enforcement, and others as applicable. The SERP members review the licensee's deficient performance, the safety significance of the finding, and any applicable regulatory requirements that should be cited.

The staff reviewed each of the examples you provided, as well as several other recent flood-related findings, and determined that although the details varied based on specific circumstances at each site, the bases for citing the violations against Appendix B to 10 CFR Part 50 were justified and adequately documented, as further discussed below. Each of the noted examples underwent a review and approval process, and those of greater significance were evaluated and dispositioned by the SERP process. The staff also noted that the licensees accepted these findings and associated violations, and initiated appropriate corrective actions to address the associated performance deficiencies.

The requirements of Appendix B to 10 CFR Part 50 apply "to all activities affecting the safety-related functions" of structures, systems, and components (SSCs). The NRC does not treat all SSCs designed to mitigate flooding at a nuclear power plant as safety-related. However, if a flood mitigation SSC is designed to protect a safety-related SSC's safety-related function during a design-basis flood, and the flood mitigation SSC would not have provided the required flood protection such that the safety-related function of the safety-related SSC would be affected during a design basis flood, then Appendix B would be applicable. Based on our review of issued flooding findings, there has been no change in a regulatory position or interpretation, so the flooding examples that you cited do not meet the definition of "backfitting" found in 10 CFR 50.109.

Thank you for your interest in this issue. The NRC will continue to make enhancements to our guidance and/or training based on feedback from all stakeholders to ensure a consistent and predictable application of our regulations. We take our safety mission and regulatory responsibilities seriously, and will continue to do so within the bounds of our lawful authority.

If you have any further questions or concerns regarding this matter, please contact Mr. Ronald Frahm at (301) 415-2986, or at ronald.frahm@nrc.gov.

Sincerely,

/RA AHowe for/

Scott Morris, Director
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission

RESPONSE TO LICENSEE STAFF'S COMMENTS

In addition to providing the exception to the Notice of Violation, the licensee also provided clarifying questions/comments via email. The NRC docketed this correspondence in ADAMS as ML15079A381 to make the information publically available. The questions/comments and the NRC responses are discussed below:

Comment 1

Comment 1 stated that the classification of the Unit 1 decay heat vault drain valves ABS-13 and ABS-14 as non-safety related is consistent with the licensing basis. These valves are maintained closed and are in the preventative maintenance program, which provides inspection on a periodic frequency. This ensures component reliability is maintained.

NRC Evaluation of Comment 1

This comment reiterated the licensee's position that the decay heat vault drain valves should not be safety-related. The NRC concluded that the non-safety-related classification of the decay heat vault drain valves is not consistent with the licensing basis function for those valves since the licensee did not address all of the required functions. The NRC determined that the flooding protection function would require the drain valves to be considered important to safety, but since the valves failed to perform their flood design function to protect the safety-related functions of the safety-related low pressure injection and containment spray systems, Appendix B to 10 CFR Part 50 is applicable. Example A.e in the Notice of Violation was re-characterized to reflect this change. However, the accident dose mitigation function to preclude any potential to impact offsite and control room operator dose from leakage inside those rooms necessitated that they should be classified as safety-related. The radiological barrier function of the valves will be addressed in NRC Inspection Report 05000313/2015004 and 05000368/2015004.

Comment 2

Comment 2 concerned the credit that the NRC used for the human reliability associated with identification of which vault would be flooding. The NRC incorrectly stated that there is a single annunciator for all three vaults in Unit 2 and, therefore, given flooding in the auxiliary building, operators would be unable to confirm if one or multiple vaults were flooding. As a result, the NRC used an assumption of poor ergonomics in the SPAR-H model for human reliability analysis. Annunciator response Procedure 2203.012L, "Annunciator 2K12 Corrective Actions," provides instructions to determine which individual room is affected. The procedure directs the operator to check the back of panel 2C-14 to determine which individual vault is impacted and to dispatch an operator to the affected room to determine the cause.

NRC Evaluation of Comment 2

The ability to identify which decay heat vault room is subject to flooding is one aspect of the human reliability analysis for recovery of decay heat removal using feed to steam generators, specifically service water system recovery. The dominant inputs for the failure probability involved the ability to align the service water supply to the emergency feedwater pump suction valves before flooding in the auxiliary building caused a loss of remote operation capability. The inspectors determined that adequate time existed for operators to diagnose and align the service water system. However, a "Poor" ergonomics factor was assigned because the diagnosis and execution would be performed without previous training and operators would be required to use sections of several procedures to accomplish the lineup, and considering the likely belief by operators that the vaults would not flood since the vaults were thought to be watertight.

The inspectors reviewed the annunciator response procedure and walked down the control room indications. The inspectors agree that there are individual vault flooding alarms inside the back of panel 2C-14 in the control room, with reflash capability. However, as stated in the report, operators would not be able to be dispatched to the affected room due to auxiliary building flooding.

Comment 3

Comment 3 questioned why the NRC believes service water system pressures would be significantly lower than what was presented during the regulatory conference, or why elevating service water system pressure would be considered a complex scenario. The licensee believed that the NRC did not provide appropriate mitigation strategy/recovery action for establishing adequate service water flow to the steam generators.

Valve 2CV-1460 (known as the squeeze valve) is not required for service water system operation, but exists only to provide a slight backpressure on the service water system in order to force makeup flow to the cooling tower (main condenser cooling medium). The inspectors were provided information comparing previous service water system pressure/flow testing against simulator alignments. This information assumed normal non-vital service water system loads remained aligned, which results in greater flow and less pressure than would be available if the service water system were aligned to the accident response mode. With valve 2CV-1460 failing to open it was estimated that the service water system pressure at the emergency feedwater pump suction would be approximately 69 psig, sufficient to provide adequate flow to the steam generator to maintain level (margin of approximately 3 gpm). NRC Inspection Report 05000313/2014010 and 05000368/2014010, however, states that service water system pressure at the emergency feedwater pump suction would be on the order of 55-60 psig. The basis for this assumption is unclear.

Although the information provided indicated sufficient service water flow and pressure to maintain steam generator level, it was recognized that the estimation was based on simulator modeling and not verified via an actual hydraulic calculation. However, such a calculation was deemed not warranted based on proceduralized simplistic action available to raise service water system pressure significantly.

NRC Evaluation of Comment 3

As stated in NRC Inspection Report 05000313/2014010 and 05000368/2014010, the inspectors used actual plant data to determine the possible service water system pressures. The data used was from actual plant events where the service water system alignment was similar to that expected during a potential flood. The inspectors also gave credit for increased service water pump discharge pressure due to flooding in Lake Dardanelle. The licensee used simulator and testing data from the service water system aligned to the emergency cooling pond, which is not representative of flooding conditions. The inspectors concluded that the data used in the inspection report was more realistic than the simulator and testing data.

The licensee commented that it was unclear why elevating the service water system pressure would be considered a complex scenario. NRC Inspection Report 05000313/2014010 and 05000368/2014010 provides a detailed explanation of the factors that resulted in the NRC's decision to consider this a complex scenario. Some of these factors include:

- The NRC determined that this recovery action would require a moderately complex diagnosis. Multiple variables would need to be evaluated including service water system alignment, unique system configurations, and pump failures in order to diagnose the lack of adequate flow to the steam generators. The ability to evaluate the service water system configuration could be impacted by flood waters throughout the buildings. No procedures existed to diagnose the need to realign valves to increase system pressure. In addition, the diagnosis would also involve re-evaluation of operator actions that were taken to align service water to emergency feedwater, since those actions did not result in feed to the steam generators as expected.

- Restoration of service water pressure to provide for service water flow to the steam generators is feasible, however, the NRC noted that procedures governing this evolution were not available to support diagnosis, the viability of the actions to restore service water system pressure had not been demonstrated or trained on, and the mitigation strategy/recovery actions had to be accomplished before the loss of natural recirculation in the reactor coolant system. Consequently, the NRC determined that there was a 29 percent failure probability for restoring service water pressure such that service water flow to the steam generators could be established. This failure probability also accounted for the dependency of the recovery diagnosis and actions on the preceding initial failure to establish sufficient service water pressure.

Comment 4

Comment 4 concerned the timing the NRC used in determining the flooding of the decay heat vaults. The amount of ingress required to initiate the decay heat vault A/B flooding alarm is approximately 1200/850 gallons, respectively. In order to obtain the timing as described in NRC Inspection Report 05000313/2014010 and 05000368/2014010 the flow-rate would have to be reduced considerably. Any reduction in flow-rate that would delay the alarms would also delay the loss of decay heat pumps P-35A/B. The increased time the components will be available directly relates to the additional analyses as it would reduce the decay heat load, the required steam generator makeup flow rate, etc. The method of recovery options would also increase and would extend the time that the decay heat pumps would be available.

Additionally, the assumptions used in the licensee's analyses were consistent. In applying modifications to the proposed inflow rates based on system resistance for the decay heat vaults and not applying the same assumptions to the general auxiliary building ingress rates, as was done in the referenced report, the allotted time before alarm and the time to achieve a water level of 335 feet in the auxiliary building general area is not conservative and would provide questionable results.

NRC Evaluation of Comment 4

As stated in NRC Inspection Report 05000313/2014010 and 05000368/2014010, the assumed inflow rate was reduced to account for conduit fill and conduit height. This would increase the time the components would be available before submergence. For external flooding, the reactor would likely be shut down for several days before floodwater could impact safety equipment resulting in a lower decay heat load, and therefore a decreased steam generator makeup rate. However, this has no impact on the probability of success for the service water system recovery scenario. As documented in the report, the inspectors concluded that there may not be forward flow of service water through the emergency feedwater system until operators took action to raise service water system pressure. If operators took action, it was assumed that adequate flow was established. Therefore, the decay heat load was not a factor in the conclusion.

The inspectors also adjusted the assumed inflow rate to the auxiliary building based on conduit fill to ensure consistent results. The conduit height assumption was not applicable to the auxiliary building inflow because the turbine building was assumed to quickly flood and, therefore, any high point in the turbine building would not delay inflow to the auxiliary building. Therefore, the inspectors applied consistent fill rate assumptions to the decay heat vaults and the auxiliary building. For all analyzed timelines, the NRC used the inspectors' vault inflow analysis, which increased the amount of time available for recovery credit.

NRC Inspection Report 2014010 documented that there was adequate time for Unit 2 operators to diagnose and align the service water system to emergency feedwater for recovery credit. However, the potential Unit 2 flow diversion, incomplete procedures, and environmental conditions were a more significant contributor to the NRC's conclusion than the timing analysis. The report also documents that there was insufficient time for Unit 1 operators to diagnose and align the service water system to emergency feedwater for recovery credit. For Unit 1, the timing analysis was not as significant to the conclusion as the environmental condition. Consequently, the proposed Unit 1 recovery received a lower failure probability and further sensitivity analysis was performed that bounded the range of assumptions.

Comment 5

Comment 5 discussed the time delay the NRC used for the availability of the alternative mitigation pump to provide water to feed both Units 1 and 2 steam generators, and the potential unavailability as a result of submerging electrical components during transportation. Procedure OP-1203.48, "Security Event," Attachment J, Section 10, prescribes how to transport and use the alternative mitigation pump to supply water to feed both Units 1 and 2 steam generators.

Using the procedure, the alternative mitigation pump and equipment trailer were timed from the secondary operations support center parking lot to the protected area (sally port), as documented in Condition Report CR-ANO-C-2014-02804. The secondary operations support center was used as a starting point due to procedural guidance that directs locating the equipment to higher elevations, and the secondary operations support center would be manned due to site flooding. The travel time required for all normal security checks was included to account for any slower travel that would occur due to postulated flood waters. The time validation to greater than 200 gpm feed to the steam generators was validated to be less than one hour for each unit and included an additional 15 minutes added for pump starting, charging fire water hoses, and manipulating plant valves to send water to the steam generators.

In addition, follow-up information was provided to the NRC regarding transportation of the normal trailer through flood waters. Movement of the alternative mitigation pump was judged to be practical using normal means (towing) up to a water level of 356 feet. Specific components that would be wetted or submerged during the fording event were evaluated and no impact due to submergence would be expected.

NRC Evaluation of Comment 5

The inspectors observed the time validation of the alternative mitigation pump strategy and reviewed Condition Report CR-ANO-C-2014-02804. The inspectors noted that the condition report, as well as the text of Comment 5, state that "the travel time required for all normal security checks was included to account for any slower travel that would occur due to postulated flood waters." The inspectors concluded that the time required to perform normal security checks does not directly correspond to the time needed to transport this equipment during the postulated flood conditions. Therefore, the inspectors evaluated the effects specific to flooding.

The inspectors concluded that, based on the water level expected to be covering the road (several feet), the probability of successfully transporting the pump on a trailer without a conveyer system of some sort anchored at each end would be low due to the forces of the water acting on the trailer.

NRC Inspection Report 05000313/2014010 and 05000368/2014010 documents that the NRC concluded that the licensee could likely take several hours to load the alternative mitigation pump onto another trailer in order to avoid submerging the pump during transport. Whether the licensee chose to test and evaluate the fording strategy, or load the pump onto a taller trailer, the result would be a several hour delay. Therefore, the licensee's comment would not change the conclusion.

Comment 6

Comment 6 stated that the NRC used incorrect failure frequencies for Units 1 and 2 circulating water system expansion joints. Unit 1 has rubber expansion joints which have a higher rupture frequency than Unit 2, which has metal expansion joints, based on EPRI data on pipe rupture frequencies. The initiating event frequency for internal flooding for Unit 1 was not provided since the consequence was insignificant, which was based on determination of flood water hydraulics. The frequency and consequence are the basis for change in risk, and since the consequence is minimal, the risk is minimal. The circulating water drains to the lake for Unit 1.

NRC Evaluation of Comment 6

NRC Inspection Report 05000313/2014010 and 05000368/2014010 documents that the licensee's evaluation of the frequency of internal flooding due to a circulating water system failure did not account for expansion joints, which are more likely to fail than piping. It was the NRC's understanding that Unit 1 contained metallic circulating water expansion joints. The NRC agrees that rubber expansion joints are expected to have a higher failure rate than metal joints. This information changes the NRC's understanding of which unit contains metallic and which unit contains rubber expansion joints. However, in NRC Inspection Report 05000313/2014009 and 05000368/2014009 (ML14253A122), the NRC considered the correct failure frequency for each unit as provided by the licensee, Unit 1 was minimal and Unit 2 was 9.03×10^{-5}/year, respectively. Therefore, this is an administrative error as the correct failure frequency for each unit was used. As a result, there is no change to the Significance Determination Process conclusions or the color of the findings.

Comment 7

Comment 7 concerned the NRC credit given for placing both units' steam generators to be in a wet layup condition during an external flooding event.

The basis for the Unit 1 steam generators to be in a wet layup condition during an external flooding event was provided in a position paper to the NRC. First, Steps 12 and 13 of Procedure 1203.025, "Natural Emergencies" requires the removal of equipment from service AND de-energizing power supplies to below-grade equipment prior to flooding, and securing of nonessential loads prior to flood waters exceeding elevation 354 feet. Second, the completion of these steps will require securing the condensate pumps and condenser vacuum pumps located below grade in the turbine building basement. Lastly, for chemistry control with the secondary system secured, steam generators will be placed in a wet layup condition in accordance with Procedures OP-1102.010, "Plant Shutdown and Cooldown," Step 11.9.3 and OP-1106.008, "OTSG Secondary Fill, Drain, and Layup."

NRC Evaluation of Comment 7

The licensee did not provide a formal position paper that was reviewed, approved, or entered into a formal records system to the inspectors. The licensee provided the inspectors with expected secondary conditions for a flood based on existing plant procedures. The discussion with the inspectors focused on the same procedures and steps that are outlined in Comment 7. No new information was provided.

As documented in NRC Inspection Report 05000313/2014010 and 05000368/2014010, the inspectors interviewed the operations managers for both units regarding the implementation of the stated procedures. The managers stated that there was some probability that the units may not be placed in wet layup given the amount of temporary flood protections, the amount of pre-planning and notice given, and the potential desire to shorten the forced outage. Additionally, NRC Inspection Report 05000313/2014010 and 05000368/2014010 documents that the NRC did not explicitly use the shorter timeline associated with the steam generators not being in a wet layup condition. If the NRC had included that assumption, it would result in additional risk to the qualitative assessment. The NRC assumed the units would be in wet layup, and this was considered to be a qualitative factor only.