UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

March 12, 2013

EA-13-018

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT - NRC INSPECTION REPORT 05000390/2012009; PRELIMINARY YELLOW FINDINGS, PRELIMINARY WHITE FINDING AND APPARENT VIOLATIONS

Dear Mr. Shea:

On February 15, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant Unit 1. The enclosed inspection report documents the inspection results which were discussed on February 21, 2013, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The enclosed inspection report discusses three findings with four Apparent Violations (AVs) associated with the site flood mitigation strategy and design control. Additional information regarding the basis for the NRC staff's significance determination is provided as an attachment to this letter. The three findings and their related AVs were evaluated using the NRC Reactor Oversight Process (ROP). One AV was evaluated using the NRC Traditional Enforcement Process.

The first finding preliminarily has been determined to be a Yellow finding with substantial safety significance that may require additional NRC inspections. As described in the enclosed report, the finding involved the failure to establish and/or maintain an Abnormal Operating Instruction procedure to mitigate onsite the effects of a probable maximum flood event. Specifically, Abnormal Operating Instruction (AOI) 7.1 "Maximum Probable Flood," Revision 21" was

~~Enclosures transmitted herewith contain SUNSI. When separated from Enclosures 2, 3, and 4 this document is decontrolled.~~

inadequate to mitigate the effects of a Probable Maximum Flood (PMF) event, in that, prior to September 30, 2009, earthen dams located upstream of the facility could potentially overtop, causing a subsequent breach. Failure of the earthen dams during a PMF event would have resulted in onsite flooding and subsequent submergence of critical equipment, such as the Emergency Diesel Generators, resulting in an ineffective flood mitigation strategy for these PMF events. This issue was assessed based on the best available information, using the applicable Significance Determination Process (SDP) in accordance with Inspection Manual Chapter (IMC) 0609, Appendix M. Following the initial review of this matter using preliminary quantitative analysis, Appendix M was used considering the uncertainties in the bounding analysis and the insights from the qualitative review. There is a lack of quantitative data and probabilistic risk assessment tools to accurately assess the risk significance of the performance deficiency in a timely manner. We also understand that there is no immediate safety concern because compensatory measures have been in place since September 30, 2009, to address this degraded condition and preclude earthen dam overtopping.

The second finding preliminarily has been determined to be a Yellow finding with substantial safety significance that may require additional NRC inspections. As described in the enclosed report, the finding involved the failure to establish and/or maintain an AOI procedure for the plant to be reconfigured and systems realigned within 27 hours of notification of a significant flooding event, consistent with Technical Requirements Manual (TRM) 3.7.2 and Watts Bar UFSAR Section 2.4. Specifically, the licensee was initially unable to implement AOI-7.1 to reconfigure and realign systems necessary for flood mitigation within 27 hours. This was based on actual walk down information associated with AOI implementation, identified spool piece fit up issues, inability to locate staged equipment, and, in general, lack of thorough understanding of the collective workload, work flow, and manpower requirements for completing flood preparation tasks. As a result, the licensee's flood mitigation strategy for certain design basis flooding events, including PMF events, was inadequate. This is not an immediate safety concern because the licensee has demonstrated an adequate capability to implement their flood mitigation strategy through procedural and process improvements.

The third finding preliminarily has been determined to be a White finding with low to moderate safety significance that may require additional NRC inspections. As described in the enclosed report, the finding involved a failure to correctly translate the design basis related to onsite flooding into the instructions for plant design change Temporary Alteration Control Form (TACF) 1-09-0006-070 Revision 2. Specifically, plant design change TACF 1-09-0006-070 for the Thermal Barrier Booster Pumps (TBBP) flood protection barrier was inadequate because: (1) the appropriate RTV sealant type was not specified in the TACF; (2) the lack of specificity regarding preparation/cleaning of surfaces as recommended by the sealant manufacturer prior to applying the sealant; (3) the failure to perform load calculations for panel deformation; and (4) failure to include provisions in the design and installation to support the temporary panels to resist deflection from hydrostatic pressure/force and potential uplift forces. As a result, the TBBP flood barrier would have failed during a probable maximum flood event, thereby submerging the TBBPs and rendering the equipment inoperable. Without the TBBPs, the probability for a Reactor Coolant Pump (RCP) seal loss of coolant accident increases, coincident with the flooding event. Currently, the licensee has re-designed the TACF and installed the barrier in the plant such that the barrier is now considered adequate to maintain the function of the TBBPs during a design basis flooding event.

All of the findings previously discussed also represent apparent violations of NRC requirements and are being considered for escalated enforcement action in accordance with the NRC Enforcement Policy which can be found on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

In accordance with NRC IMC 0609, Significance Determination Process, we intend to complete our risk evaluations using the best available information and issue our final significance determination within 90 days of the date of this letter. The Significance Determination Process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination. Before the NRC makes its final decision on this matter, we are providing you an opportunity to either: (1) present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at these findings and their significance at a Regulatory Conference, or (2) submit your position on these findings to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference. If you decide to submit only a written response, such a submittal should be sent to the NRC within 30 days of the receipt of this letter. If you decline to request either a Regulatory Conference or submit a written response, you relinquish your right to appeal the final significance determination; in that, by not doing either you fail to meet the appeal requirements stated in the Prerequisites and Limitations sections of Attachment 2 of IMC 0609.

One additional AV associated with the first preliminarily Yellow finding is also being considered for escalated enforcement action in accordance with the NRC Traditional Enforcement Policy. Specifically, this issue involved the failure to report an unanalyzed condition, as required by 10 CFR 50.72. The licensee has had compensatory actions in place since September 30, 2009, for this condition and has since reported the unanalyzed condition to the NRC on February 6, 2013. This AV is being evaluated using the NRC's enforcement process because it impacted NRC's ability to perform its regulatory function and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. Additional details for this AV are provided in the enclosed inspection report.

Before the NRC makes its enforcement decision, we are providing you an opportunity to respond to this AV addressed in this inspection report within 30 days of the date of this letter, or request a Pre-decisional Enforcement Conference (PEC). If a PEC is held, it will be open for public observation.

If you choose to provide a written response, it should be clearly marked as "Response to Apparent Violation in Inspection Report No. 05000390/2012009"; EA-13-018, and should include for the apparent violation: the reason for the apparent violation, or, if contested, the basis for disputing the apparent violation; the corrective steps that have been taken and the results achieved; the corrective steps that will be taken to avoid further violations; and the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. If an adequate response is not received within the time specified or an extension of time has not been granted by the NRC, the NRC will proceed with its enforcement decision.

If you choose to request a PEC, the conference will afford you the opportunity to provide your perspective on the apparent violation and any other information you believe the NRC should take into consideration before making an enforcement decision. The topics discussed during the conference may include the following: information to determine whether the violation occurred, information to determine the significance of the violation, information related to the identification of the violation, and information related to any corrective actions taken or planned to be taken. In presenting your corrective actions, you should be aware that the promptness and comprehensiveness of your actions will be considered in assessing any civil penalty for the apparent violation.

Should you choose to request a Regulatory Conference/REC, a joint conference may be appropriate based on the commonality of these identified issues.

Please contact Scott Shaeffer at (404) 997-4521 within 10 days of the date of this letter to notify the NRC of your intended response. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Since the NRC has not made a final determination as to the significance of these issues, no Notice of Violation is being issued at this time. Please be advised that the number and characterization of the apparent violations described in the enclosure may change as a result of further NRC review. You will be advised by separate correspondence of the results of our deliberations on this matter.

Additionally, one NRC identified and one licensee identified finding of very low safety significance (Green) were identified during this inspection. These findings were determined to involve a violation of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Watts Bar Unit 1 facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, 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. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

Richard P. Croteau, Director
Division of Reactor Projects

Docket No. 50-390
License No. NPF-90

Enclosures:
1. NRC Inspection Report 05000390/2012009
   w/Attachment: Supplemental Information
2. Phase 3 w/Attachments: Failure to Ensure Onsite
   Electrical Power During a PMF Event **(OFFICAL USE ONLY – SECURITY RELATED INFORMATION)**
3. Phase 3 w/Attachments: Failure to Maintain an
   Adequate Abnormal Condition Procedure to
   Implement the Flood Mitigation Strategy **(OFFICAL USE ONLY – SECURITY RELATED INFORMATION)**
4. Phase 3 w/Attachments: Inadequate Design of
   Thermal Barrier Booster Pumps' Flood Protection Barrier **(OFFICAL USE ONLY – SECURITY RELATED INFORMATION)**

cc: (See page 6)

cc w/o encl:

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-390
License No.:	NPF-90
Report No.:	05000390/2012009
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Unit 1
Location:	Spring City, TN 37381
Dates:	October 7, 2012, through February 15, 2013
Inspectors:	R. Monk, Senior Resident Inspector K. Miller, Resident Inspector
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000390/2012009; 10/7/2012 – 02/15/2013; Watts Bar Nuclear Plant, Unit 1; Adverse Weather Protection.

The report covers a period of inspection by resident inspectors. One Non-Cited Violation, one Licensee identified violation of very low safety significance (Green), and four Apparent Violations were identified. The significance of the appropriate findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

TBD: The inspectors identified an Apparent Violation (AV) of Technical Specification (TS) 5.7.1, Procedures, for the licensee’s failure to establish an adequate procedure for mitigation of external events, specifically flooding prior to the installation of HESCO barriers in 2009. The inspectors determined that the licensee’s failure to comply with TS 5.7.1, “Procedures,” was a performance deficiency. Specifically, AOI-7.1, Maximum Probable Flood, was not adequate to prevent the loss of critical safety functions (e.g., emergency power) during a PMF event prior to the installation of the HESCO barriers and other compensatory measures. This procedure, in part, is used to maintain the established license basis for compliance with 10 CFR 50, Appendix A, General Design Criteria 2, “Design Bases for Protection Against Natural Phenomena.” Failure to establish adequate procedures for flood mitigation results in a failure to maintain adequate protection against natural phenomena in accordance with the licensing basis of the plant.

This performance deficiency was considered more than minor because it was associated with the protection against external factors attribute of the Reactor Safety/ Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, prior to the installation of the HESCO barriers and other compensatory measures, Abnormal Conditions procedure, AOI 7.1, “Maximum Probable Flood,” was not adequate to prevent the loss of Emergency Power during a PMF event. Loss of emergency power would lead to core and/or spent fuel pool inventory damage. The combination of the event frequencies and types of rainfall events which would over-top earthen dams leading to the loss of emergency power resulting in core damage has an impact of substantial safety significance. The NRC concluded that the significance of the finding is preliminarily of substantial safety significance (Yellow). The inspectors determined that no cross-cutting aspect was applicable. (Section 1R01.1)

TBD: The inspectors identified an AV of 10 CFR 50.72(b)(3)(ii)(B), "Immediate Notification Requirements for Operating Nuclear Reactors," for failure to report within eight hours an unanalyzed condition that significantly degraded plant safety. Specifically, the licensee failed to notify the NRC upon discovery that a postulated PMF would result in the overtopping of earthen dams not previously assumed in the plant design. The failure to report this unanalyzed condition resulted in the NRC not being made aware of a condition which would have resulted in additional NRC review. Specifically, the failure to notify the NRC within eight hours of discovery of an unanalyzed condition that significantly degraded plant safety and resulted in an unacceptable change to the facility or procedures. The inspectors determined an evaluation for cross-cutting aspect was not applicable because this is a traditional enforcement violation. (Section 1R01.2)

TBD: The inspectors identified an AV of Technical Specification 5.7.1, Procedures, for the licensee's inability to demonstrate that the required Stage I and Stage II activities could be performed within 27 hours as required by AOI-7.1, Maximum Probable Flood. The licensee's failure to adequately demonstrate the ability to realign plant systems into their flood mode configuration using AOI-7.1, Maximum Probable Flood, within the time frame required by TRM 3.7.2 and Watts Bar UFSAR Section 2.4, which could directly lead to the inability to remove decay heat from the reactor core resulting in core damage, was a performance deficiency. This performance deficiency was considered more than minor because it was associated with the Protection Against External Factors attribute of the Reactor Safety/ Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the inability to realign plant systems into their flood mode configuration within the required time frame could directly lead to the inability to remove decay heat. The combination of the seismic and rainfall event frequencies and types of rainfall events which would lead to flooding above site grade and the inability to realign plant systems into their flood mode configuration within the 27-hour required time frame could directly lead to the inability to remove decay heat from the reactor core resulting in core damage which has an impact of substantial safety significance. The NRC concluded that the significance of the finding is preliminarily substantial safety significance (Yellow).

The cause of the finding had a cross-cutting component of Resources in the area of Human Performance with an aspect of ensuring that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, inadequacies in those procedures, equipment, and personnel training necessary to realign plant systems within the required time frame to cope with all anticipated external flooding events. (H.2 (d)) (Section 1R01.3)

TBD: The inspectors identified an AV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to adequately protect safety-related equipment during flood mode preparation. The licensee’s failure to adequately protect safety-related equipment during flood mode preparation as implemented by AOI-7.1, Maximum Probable Flood, was a performance deficiency. This performance deficiency was considered more than minor because it was associated with the Protection Against External Factors attribute of the Reactor Safety/Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the TACF was incapable of preventing water damage, during a PMF event, to both trains of important equipment, specifically the thermal barrier booster pumps (TBBPs), necessary for coping with the PMF impact on Unit 1. Without the TBBPs and with reactor coolant pump (RCP) seal injection lost, there is no engineering assurance that RCP seal damage would not occur, leading to an RCP seal loss of coolant accident (LOCA).

The performance deficiency involved external events. Consequently a Phase 2 analysis could not be performed and therefore a Phase 3 analysis was conducted. The increase in core damage frequency (ΔCDF) for this issue was estimated at 6.35×10^{-6}; which has an impact of low to moderate safety significance. The NRC concluded that the significance of the finding is preliminarily of low to moderate safety significance (White).

This finding has a cross-cutting aspect in the Work Practices component of the Human Performance area because it was directly related to the licensee not ensuring adequate supervisory and management oversight of engineering design work activities associated with a plant design change to protect the TBBPs during certain flood events. (H.4 (c)). (Section 1R01.4)

Green: The inspectors identified two examples of an NCV of the 10 CFR 50 Criteria XVI, “Corrective Action,” for failure to correct conditions adverse to quality for the intake pumping station (IPS) CKV-040-0604, pump 3B, discharge check valve which resulted in it being non-functional for an extended period of time, and both the IPS 3A and 3B sump pumps, which resulted in the pumps remaining in a degraded condition for an extended period of time. The licensee’s failure to maintain these components in accordance with the requirements of the augmented in-service testing program and WB-DC-40-29, Flood Protection Provisions, were performance deficiencies. The performance deficiencies were determined to be more than minor because, if left uncorrected, would lead to a more significant safety concern. Specifically, internal flooding of the IPS mechanical equipment room housing the train A essential raw cooling water (ERCW) strainers could occur. The inspectors performed a Phase 1 evaluation per IMC 0609, Attachment 4, and determined that the finding was potentially risk significant because it involved the degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors). Consequently a Phase 3 analysis was performed by a Senior Reactor Analyst. The analyst determined the finding was of very low safety significance, Green. These findings directly involved the cross-cutting area of Human Performance under the Work

Practices component, in that, the licensee failed to provide adequate supervisory and management oversight to ensure corrective actions were taken to maintain the functionality of IPS equipment for extended periods of time. (H.4 (c)) (Section 1R01.5)

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program (CAP). That violation and corrective action tracking number are listed in Section 40A7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 External Flood Protection Inspection

a. Inspection Scope

The inspectors reviewed the licensee’s readiness to cope with external flooding. External flooding from a probable maximum flood (PMF) or design basis flood (DBF) has the potential to cause internal flooding of a portion of a number of the plant structures. During this type of external flooding event, the reactor core decay heat will be removed by the flood protection provisions designed to remain operational up to the DBF elevation and maintain compliance with General Design Criterion 2 regarding external flooding. Provisions have also been made to cool the spent fuel pool. Abnormal Operating Instruction (AOI)-7.01 “Maximum Probable Flood,” Revision 21 documents the realignment and shutdown requirements for the plant during this event. The inspectors reviewed the feasibility of several of these provisions for coping with this type of event to determine if desired results would be achieved. The inspectors also reviewed the licensee’s related corrective action documents (problem evaluation reports) to ensure any nonconforming conditions related to potential flooding were properly addressed. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample.

b. Findings

.1 Inadequate Abnormal Condition Procedure for Flood Mitigation Strategy Prior to Installation of HESCO Barriers

Introduction: The inspectors identified an Apparent Violation (AV) of Technical Specification (TS) 5.7.1,” Procedures,” for the licensee’s failure to establish an adequate procedure for mitigation of external events, specifically flooding prior to the installation of HESCO barriers. HESCO barriers are engineered wire baskets lined with a fabric material and loaded with crushed gravel. Individual baskets are interconnected to form a boundary used to prevent over wash of earthen embankments.

Description: In February 2008, NRC performed a quality assurance (QA) inspection of the flood-related combined license application (COLA) submittal information for Bellefonte Nuclear Plant (BLN) Units 3/4. In the course of the QA inspection, NRC reviewed a 1998 calculation performed for the TVA operating units to evaluate the effects of physical changes resulting from the National Dam Safety program to the reservoir system on the plant design basis flood calculations. NRC identified that the 1998 calculation did not meet the TVA procedural requirement in place at that time with respect to verification and validation of the software and documentation and verification of the input parameters required for those analyses. Consequently, TVA initiated Problem Evaluation Report (PER) 138749 during the course of that inspection. A Notice of Violation (NOV) was issued on March 19, 2008, against the BLN 3/4 COLA submittal for that plant's use of the 1998 calculation.

PER 138749 was written to document and evaluate impacts to operating plants throughout the process of bringing the software and design inputs under configuration in accordance with Nuclear Power Group's (NPG) QA Plan. Each operating site also initiated a PER to confirm continued functionality. TVA began validating and verifying the codes and inputs associated with PMF calculations.

The corrective actions for PER 138749 included a process for the identification and evaluation of "anomalies" in the course of the re-verification process. The evaluation of these anomalies included review and signoffs on an anomaly documentation form by personnel from TVA's River Operations and NPG. The anomalies were categorized as either enhancements or errors. Cumulative effects of all PERs were tracked in PER 138749, which also contained a table that described each anomaly, summarized the evaluation of the error, and tabulated the site and corporate PER numbers.

On July 28, 2009, the licensee determined that the spillway discharge coefficient previously used in the Fort Loudoun Dam rating curve was inconsistent with more recent model test data. Correcting this discharge coefficient resulted in less flow through the Fort Loudoun Dam spillways at the high headwater elevation during a PMF and would potentially over-top earthen portions of the Fort Loudoun Dam. Failure of the dam was assumed if the earthen portion over-topped. Based on these results, the licensee documented that the PMF levels were expected to exceed the original design licensing basis elevations of 738.1 feet, and 722.6 feet at Watts Bar and Sequoyah Nuclear Plants, respectively.

Inspectors, in that time frame, were informed that there were uncertainties in the PMF levels and, at that point, a bulldozer would be placed at Fort Loudoun Dam to cut a temporary channel in the marina saddle dam to prevent embankment erosion of the dam. This was communicated as a precautionary measure and indicated that would be in place from September 30, 2009, until December 31, 2009. PER 177501 documented this condition. The associated functional evaluation for this PER, Technical Basis for Functionality, Revision (Rev.) 1, Fort Loudoun Dam Spillway Coefficient, states the following as a conclusion:

> This assessment covers only the time period between June and December. Based on the PMF level analysis performed with the Simulated Open Channel Hydraulics (SOCH), using appropriate inputs for seasonal rain-runoff, and taking credit for the compensatory action listed above, it is concluded that Cherokee, Fort Loudoun, Tellico, and Watts Bar dams will not overtop, and the original design basis PMF levels at BFN, SQN, and WBN of 572.5, 722.6, and 738.1 feet respectively will not be exceeded.

On December 30, 2009, calculation CDQ000020080054, Rev. 0, PMF Determination for Tennessee River Watershed, was issued. Shortly following release, inspectors were informed by the licensee that the calculation for the new PMF levels was complete and that the PMF level for Watts Bar Nuclear Plant had increased from the original licensing basis number of 738.1 feet to 738.8 feet. No mention was made of the need to continue the previous precautionary measure for Fort Loudoun Dam or the need for additional or different precautionary measures for over-topping.

Subsequent review of this calculation by the inspectors did not indicate the need for any type of temporary measures to protect any of the four affected dams, Cherokee, Fort Loudoun, Tellico, or Watts Bar. The temporary HESCO barriers are credited in the above calculation. These barriers are interlocking 15'x3'x3' baskets filled with finely crushed gravel which, in effect, raises the height of the dam. However, inspectors did not find any reference to these temporary barriers in their review. According to the licensee's response to the NRC Confirmatory Action Letter dated October 30, 2012, failure of the HESCO barriers coincident with a PMF event would place the licensee outside their design PMF basis. Subsequent licensee analysis as part of the development of calculation CDQ000020080054, Rev. 0, PMF Determination for Tennessee River Watershed, also confirmed that the issue related to the non-conservative Fort Loudoun Dam spillway coefficients existed prior to the original plant licensing.

Analysis: The inspectors determined that the licensee's failure to comply with TS 5.7.1, "Procedures," was a performance deficiency. Specifically, AOI-7.1, Maximum Probable Flood, was not adequate to prevent the loss of critical safety functions (e.g., emergency power) during a PMF event prior to the installation of the HESCO barriers and other compensatory measures. This procedure, in part, is used to maintain the established license basis for compliance with 10 CFR 50, Appendix A, General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena." Failure to establish adequate procedures for flood mitigation results in a failure to maintain adequate protection against natural phenomena in accordance with the licensing basis of the plant.

This performance deficiency was considered more than minor because it was associated with the protection against external factors attribute of the Reactor Safety/Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, prior to the installation of the HESCO barriers and other compensatory measures, Abnormal Conditions procedure, AOI 7.1, "Maximum Probable Flood," was not adequate to prevent the loss of Emergency Power during a PMF event. Loss of emergency power would lead to core damage.

An adequate surrogate for the loss of all equipment due to submergence was not needed for Watts Bar Unit 1 (Conditional Core Damage Probability = 1.0 given dam failure). Further, the performance deficiency involved external events. Consequently a Phase 2 analysis could not be performed and the issue went directly to a Phase 3 analysis.

A Senior Reactor Analyst performed a Phase III evaluation in accordance with IMC 609, “Significance Determination Process,” Appendix M, and determined that treatment of this issue as a Yellow finding primarily based on the assumed event frequency without intervening mitigation. In addition, the analyst determined that there was a population of rainfall events (of less severity and greater frequency than the Probable Maximum Precipitation event) that could cause overtopping of upstream earthen dams, and hence would potentially add to the risk significance of the issue. The NRC concluded that potential over-topping of earthen dams leading to the loss of emergency power resulting in core damage is preliminarily of substantial safety significance (Yellow). The cause of the finding extends back through all procedure revisions prior to 2009. Therefore, it is not related to current performance and is not assigned a cross-cutting aspect. For the complete analysis, see Enclosure 2 of this inspection report.

The inspectors did not identify a cross-cutting aspect associated with this finding because the performance deficiency extends back through all procedure revisions prior to 2009 and does not represent current licensee performance.

<u>Enforcement</u>: Technical Specification 5.7.1, Procedures, requires in part that written procedures shall be established, implemented, and maintained covering the following activities: The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; Appendix A, Section 5, requires procedures for Abnormal Conditions.

Abnormal Operating Instruction (AOI) 7.1, “Maximum Probable Flood,” Revision 21, provides detailed instructions for implementing required site flood mitigation strategies necessary to cope with design basis flooding events.

Contrary to the above, prior to September 30, 2009, the licensee failed to establish an adequate Abnormal Operating Instruction procedure to implement its flood mitigation strategy. Specifically, AOI 7.1 was inadequate to mitigate the effects of a Probable Maximum Flood (PMF) event, in that earthen dams located upstream of the facility could potentially overtop, causing a subsequent breach. Failure of the earthen dams during a PMF event would have resulted in onsite flooding and subsequent submergence of critical equipment, such as the Emergency Diesel Generators, resulting in an ineffective flood mitigation strategy for these PMF events. This violation existed since the initial operating license until compensatory measures were put in place to prevent over-topping of the earthen portions of the Ft. Loudoun Dam. This issue was entered into their corrective action program as PER 682212. This violation is being treated as an AV, consistent with Section 2.3.3 of the NRC Enforcement Policy and is identified as AV 05000390/2012009-01: Inadequate Abnormal Condition Procedure for Flood Mitigation Strategy Prior to Installation of HESCO Barriers.

.2 <u>Failure to Report Unanalyzed Condition Related to External Flooding</u>

<u>Introduction</u>: The inspectors identified an AV of 10 CFR 50.72(b)(3)(ii)(B), "Immediate Notification Requirements for Operating Nuclear Reactors," for failure to report within eight hours an unanalyzed condition that significantly degraded plant safety.

Specifically, the licensee failed to notify the NRC upon confirmation that a postulated PMF event would result in overtopping of critical earthen dam structures upstream of the facility. Subsequent analysis identified this condition would have adversely impacted operability of all emergency diesel generators.

<u>Description</u>: As a result of a previous NRC-identified NOV related to postulated flooding levels, in support of Bellefonte Nuclear licensing, the New Generation Design and Construction (NGDC) organization initiated PER 138749. This PER was to document and evaluate impacts to operating plants throughout the process of bringing the software and design inputs under configuration in accordance with NPG's QA Plan. Each operating site also initiated a PER to confirm continued functionality. TVA began validating and verifying the codes and inputs associated with PMF calculations.

In recent review of documents associated with this PER, the inspectors determined that the licensee had documented, on or about July 28, 2009, in PER 177492 from NGDC that due to potential incorrect flow coefficients on the Fort Loudoun Dam, reservoir levels would exceed the height of the dam. The consequences were documented that "…PMF levels are expected to exceed the original design and licensing basis elevations of 738.1, 722.6, and 572.5 at WBN, SQN, and BFN, respectively" (units are feet above sea level). Similar statements were documented about the Watts Bar, Tellico, and Cherokee dams. For additional details, see AV 0500390/2012009-01 contained within this report.

As a result of PER 177492, four additional PERs were generated by TVA Nuclear Corporate (PER 177501), WBN (PER 177669), SQN (PER 177822), and Browns Ferry Nuclear Plant [BFN] (PER 178130), respectively, for the Ft. Loudoun incorrect flow coefficient issue. The corporate PER was marked as 'Not Reportable'. Each of the plant site PERs were marked as 'Potentially Reportable'. The site licensing review by each one of the respective sites was never completed. Similarly, PERs were generated for corporate and the sites for each of the other three dams. All of these PERs relied on one functional evaluation for corporate PER 177501. This functional evaluation was completed on September 30, 2009. In essence, it said that the original licensing basis of all the plant sites would be maintained with a compensatory measure to change the TVA River Operations organization flooding notification to a rain event of 8.5 inches in 7 days and the removal of the Ft. Loudoun marina saddle dam with a bulldozer should the flooding conditions of concern be expected. This functional evaluation was issued on September 30, 2009, with an expiration date of December 31, 2009.

On September 30, 2009, Sequoyah Nuclear Plant (SQN) issued event number 45395 making a non-emergency report due to TVA notifying various government agencies and the media that TVA intended to place some temporary structures on Ft. Loudoun, Cherokee, Tellico, and Watts Bar dams for the purpose of raising the height of the dams. This was characterized as a precautionary measure to prevent possible over-topping of the dams in the event of a hypothetical extreme flooding event. These precautionary measures were stated to ensure that Watts Bar remained within its original licensing basis. This report did not characterize the information as an unanalyzed condition and identified the proposed temporary structures as precautionary measures.

On December 30, 2009, calculation CDQ000020080054, Rev. 0, PMF Determination for Tennessee River Watershed, was issued. Shortly following release, the inspectors were informed by the licensee that the calculation for the new PMF levels was complete and that the PMF level for the Watts Bar plant had increased from the original licensing basis number of 738.1 feet to 738.8 feet. No mention was made of the need to continue the previous precautionary measure for Fort Loudoun Dam or the need for additional or different measures for over-topping.

<u>Analysis</u>: During the current inspection period, the inspectors determined the failure to provide an eight-hour report of an unanalyzed condition that significantly degrades plant safety was contrary to 10 CFR 50 Part 50.72(b)(3)(ii)(B) and was a performance deficiency. The performance deficiency was evaluated using IMC 0612, “Power Reactor Inspection Reports,” and was determined to be of more than minor significance. However, it was also determined to involve a traditional enforcement violation because it potentially impeded or impacted the regulatory process. Specifically, failure to notify the NRC of an unanalyzed condition challenges the regulatory process because it prevents the NRC from evaluating the need to expand the scope of inspection to include the circumstances surrounding the condition. The traditional enforcement violation was determined to be more than minor in accordance with the NRC Enforcement Policy because the information that was not reported to NRC had a material impact on safety and licensed activities.

Specifically, the failure to notify the NRC within eight hours of discovery of an unanalyzed condition that significantly degraded plant safety and resulted in an unacceptable change to the facility or procedures. The inspectors determined an evaluation for cross-cutting aspect was not applicable because this is a traditional enforcement violation.

<u>Enforcement</u>: 10 CFR 50.72(b)(3)(ii)(B), “Immediate Notification Requirements for Operating Nuclear Reactors,” requires, in part, that licensee’s report, within eight hours, an unanalyzed condition that significantly degraded plant safety.

Contrary to the above, on December 30, 2009, the licensee failed to report within eight hours an unanalyzed condition that significantly degraded plant safety for the Watts Bar Unit 1 facility. Specifically, the licensee failed to notify the NRC upon confirmation that a postulated Probable Maximum Flood (PMF) event would result in overtopping of critical earthen dam structures upstream of the Watts Bar facility. These overtopping conditions were not previously assumed in the licensing basis for the facility and represented an unanalyzed condition.

When identified by the NRC, the licensee entered this into the CAP as PER 669443 and 682202. The licensee has had compensatory actions in place since September 30, 2009, for this condition and has since reported the unanalyzed condition to the NRC on February 6, 2013. The NRC's review of the impact of the unanalyzed condition prior to establishment of the compensatory actions was addressed in Section 1R01.1. This issue is identified as AV 05000390/2012009-02, Failure to Report Unanalyzed Condition Related to External Flooding.

.3 Failure to Maintain an Adequate Abnormal Condition Procedure to Implement the Flood Mitigation Strategy

Introduction: The inspectors identified an AV of Technical Specification 5.7.1, Procedures for the licensee's inability to demonstrate that the required Stage I and Stage II activities could be performed within 27 hours as required by AOI-7.1, Maximum Probable Flood.

Description: The inspectors identified that the plant could not achieve flood mode configuration in the required time. The TRM, Section 3.7.2, Flood Protection Plan, specifies communications between the licensee and the TVA River Operations organization and the time frames for these communications. Based on these communications, which are broken up into Stage I and Stage II, the licensee is required to take certain actions. The actions for Stage I are essentially preparatory in nature for the plant site to prepare to receive flooding levels above plant grade. These prepatory actions include shutting down the reactor, commencing cool down to 350 degrees F, and movement of equipment. These Stage I activities are to be complete within 10 hours of the determination that Stage I should be implemented. Based on communications with TVA River Operations, the licensee remains in Stage I until River Operations determines that flood levels may reach plant grade level. At this point, Stage II is entered where significant plant system realignments occur including cross-connecting the essential raw cooling water (ERCW) system to the component cooling system (CCS), ERCW to the raw cooling water (RCW) system, the fire protection system to the auxiliary feed water (AFW) system, and in some plant conditions, spent fuel pool (SFP) cooling to the residual heat removal (RHR) system. Stage II activities are to be completed within 17 hours. Stage II activities will likely have irrevocable consequences on plant equipment (e.g., steam generators will be significantly contaminated by raw water from fire protection). The licensee has indicated that these Stage II actions will not be entered into without a firm degree of certainty by the licensee that flood waters will reach plant grade.

Inspectors observed the licensee simulating installation of the flood mode piping spool pieces. On some of the larger spool pieces weighing upward of 400 pounds, this included connection of chain falls to non-specified rigging points, additional measures to account for piping spring back resulting in a short spool piece, and other installation issues. Based on these observations, the tools, procedures, and manpower usage did not indicate that the licensee would be successful at this plant reconfiguration effort within the 17-hour Stage II window. Initial efforts at integration of these observed maintenance procedures within the master AOI-7.1, Maximum Probable Flood, yielded a time of approximately 39 hours. Therefore, the total time of Stage I and Stage II activities would have exceeded the assumed 27 hours for successful implementation of their flood mitigation strategy.

Based on this information, the licensee utilized field input and other data to improve resource loading and sequencing of the support procedures over a three-day effort resulting in an implementation time reduction to 32 hours and 37 minutes. With additional focus based on previous field demonstration of one particular supporting procedure, Maintenance Instruction (MI)-17.021, Installation of Spool Pieces between ERCW and Component Cooling Systems, the time was further reduced to 27 hours and 34 minutes. This was accomplished by assigning two maintenance teams working in parallel on the two largest, heaviest spool pieces. A further reduction in the time requirements of AOI-7.10, Flood Mode Electrical Systems Alignment, by working parallel teams on the four electric power shutdown boards vice in series, yielded a time reduction to 25 hours and 57 minutes.

Current Watts Bar licensing basis, in accordance with UFSAR 2.4, Hydrologic Engineering, WB-DC-40-29, Flood Protection Provisions, and TRM 3.7.2, Flood Protection Plan, requires that the licensee be able to place the plant in a flood mode configuration within 27 hours from the time of a flood warning. The Watts Bar Final Safety Analysis Report (FSAR) Section 2.4.14.4.3 states the following: "The steps needed to prepare the plant for flood mode operation can be accomplished within 24 hours of notification that a flood above plant grade is expected. An additional 3 hours are available for contingency margin."

NRC observation of the table top simulations of AOI-7.1, Maximum Probable Flood, and the supporting maintenance instructions, indicated that the plant cannot be reconfigured in the allowed 27-hour time frame. The licensee was not able to initially demonstrate an acceptable capability to implement the flood protection measures in the allowable time as described in the FSAR. The ability of the licensee to perform these activities in the time allotted by the TRM was not assured given the number of days of refinement required by the licensee to reduce the time to the current 25 hours and 57 minutes.

The resident inspectors observed the required flooding walkdowns that the licensee performed. The extent of condition for this issue would extend beyond the most limiting flood (i.e., the 27-hour event) to potentially other "slower moving" events that could also exceed plant grade and submerge the lower levels of the auxiliary building. The following illustrates examples of inadequacies which would have added time to the flood mode configuration process:

- After inspectors identified piping interferences preventing installation of the 20-inch ERCW to CCS spool piece, it was relocated in a manner that allowed more direct access and ease of installation which shortened the demonstration timeline.

- The temporary wall around the thermal barrier booster pumps (TBBPs) was not installed until July 2012. It had been credited since January 2010 and would have required approximately 40 man-hours to build. This was not accounted for in the aforementioned timeline.

- MI-17.017, Flood Preparation - Drain Collector Tanks, provides steps for attaching a fire hose to the passive failure connection on the floor and equipment drain sump for filling of the floor drain collector tank 0-TNK-77-107 and the tritiated drain collector tank 0-TANK-77-2 by utilizing a 2-inch, 150-lb flange adapter with 1-1/2 inch fire hose connections. The fire hose adapter connection could not be located.

- MI-17.029 requires the use of a 2-inch pipe flange with a 2-1/2 inch male fire hose adapter. The fire hose adapter connection could not be located.

- Contrary to the requirements of the AOI-7.03, Flood Mode CVCS and WDS Tank Filling Instructions, Step 3.2.h.1 is not consistent with the plant configuration. The piping drain was both mis-labeled and not in the location specified.

Analysis: The licensee’s failure to adequately demonstrate the ability to realign plant systems into their flood mode configuration using AOI-7.1, Maximum Probable Flood, within the time frame required by TRM 3.7.2 and Watts Bar UFSAR Section 2.4, which could directly lead to the inability to remove decay heat from the reactor core resulting in core damage, was a performance deficiency. This performance deficiency was considered more than minor because it was associated with the Protection Against External Factors attribute of the Reactor Safety/Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the inability to realign plant systems into their flood mode configuration within the required time frame could directly lead to the inability to remove decay heat.

A Senior Reactor Analyst performed a Phase III evaluation in accordance with IMC 609, “Significance Determination Process,” and determined that the increase in core damage frequency (ΔCDF) for this event was approximately 1.4×10^{-5}. Treatment of this issue as a Yellow finding is also appropriate given the significant uncertainties and the fact that rainfall events which would lead to flooding above site grade and the licensee’s inability to realign plant systems into their flood mode configuration within the 27-hour required time frame could directly lead to the inability to remove decay heat from the reactor core resulting in core damage. The NRC concluded that the significance of the finding is preliminarily of substantial safety significance (Yellow). For the complete analysis, see Enclosure 3 to this inspection report.

The cause of the finding had a cross-cutting component of Resources in the area of Human Performance with an aspect of ensuring that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, those procedures, equipment, and personnel and training necessary to realign plant systems within the required time frame to cope with all anticipated external flooding events.

Enforcement: Technical Specification 5.7.1, Procedures, requires, in part, that written procedures shall be established, implemented, and maintained covering the following activities: The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; Appendix A, Section 5, requires procedures for Abnormal Conditions.

Abnormal Operating Instruction AOI-7.1, “Maximum Probable Flood,” Revision 21, and the supporting maintenance instructions referenced therein, required that the plant be reconfigured and systems realigned within 27 hours, consistent with Technical Requirements Manual (TRM) 3.7.2 and Watts Bar UFSAR Section 2.4.

Contrary to the above, the licensee failed to maintain an adequate Abnormal Operating Instruction procedure to implement its flood mitigation strategy. Specifically, the licensee was unable to implement AOI-7.1 to reconfigure and realignment systems necessary for flood mitigation within 27 hours. As a result, the licensee’s flood mitigation strategy for certain flooding events, including PMF events, was inadequate. This condition existed from initial licensing until July 2012. This issue was entered into their corrective action program as PER 635837. This violation is being treated as an AV, consistent with Section 2.3.3 of the NRC Enforcement Policy and is identified as AV 05000390/2012009-03, Failure to Maintain an Adequate Abnormal Condition Procedure to Implement the Flood Mitigation Strategy.

.4 Failure to Adequately Protect Safety-Related Equipment During Flood Mode Preparation

Introduction: The inspectors identified an AV of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to adequately protect safety-related equipment during flood mode preparation as implemented in AOI-7.1, "Maximum Probable Flood."

Description: On July 9, 2012, the inspectors identified that the licensee did not adequately design a plant modification, Temporary Alteration Control Form (TACF) 1-09-0006-070, Rev. 2, for flood mode preparation to prevent submergence of the TBBPs in a PMF event. Specifically, procedure 0-MI-17.004 provided a means to erect a temporary, water-tight flood barrier around both trains of TBBPs during a Stage I (flood) warning (10 hours). This barrier was designed to be sealed with a sealant that would not prevent water intrusion. The inspectors identified that the engineering-specified room temperature vulcanizing (RTV) sealant was labeled "not for use under water." The original revision of this TACF was established in 2009 as a compensatory measure to be installed in the event of a significant flood for increased PMF levels which were calculated by the licensee to potentially impact the TBBPs.

Additional design issues identified by the inspectors on July 14, 2012, during the partial installation of the TACF included: (1) the lack of adequate preparation/cleaning of surfaces as recommended by the sealant manufacturer prior to applying the sealant; (2) failure to perform load calculations for panel deformation; and (3) failure to include provisions in the design and installation to support the temporary panels to resist deflection from hydrostatic pressure/force and potential uplift forces, since the panels were not anchored to the floor. A subsequent calculation (CDQ 0010702012000060, Rev. 0) prepared by the licensee on July 19, 2012, confirmed approximately 1.9 inches of panel deflection due to the hydrostatic forces. This amount of panel deflection would have resulted in water intrusion past the barrier sealant applied between the panels and the floor.

On August 20 and 21, 2012, the installed panel assembly was completely removed and retained for a future testing and replaced by a new panel installation under TACF 1-09-0006-070, Rev. 5, and 0-MI-17.004, Rev. 5. This replacement activity included preparation/cleaning of surfaces to be sealed and use of an approved sealant primer, as recommended by the sealant manufacturer, prior to applying the sealant. Although not specified or documented in the latest TACF or 0-MI-17.004, a vertical brace was installed to resist potential uplift forces. On September 29, 2012, the licensee performed a full-scale test of the TACF 1-09-0006-070 R2 design with the original panels (removed on August 20, 2012) and the RTV sealant, and the test failed proving the original barrier design, with RTV and without horizontal bracing behind the panels to resist deflection from hydrostatic pressure/force, was incapable of protecting the TBBPs during a PMF event.

Analysis: The licensee's failure to adequately protect safety-related equipment during flood mode preparation in accordance with AOI-7.1 was a performance deficiency. This performance deficiency was considered more than minor because it was associated with the Protection Against External Factors attribute of the Reactor Safety/Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the TACF was incapable of preventing water damage, during a PMF event, to both trains of important equipment (TBBPs) necessary for coping with the PMF impact on Unit 1. Without the TBBPs and with RCP seal injection lost, there is no engineering assurance that RCP seal damage would not occur, leading to an RCP seal loss of coolant accident (LOCA).

An adequate surrogate for the failure of RCP seal injection while in a shutdown condition could not be found in either the Phase 2 pre-solved worksheets or the Phase 2 notebooks for Watts Bar Unit 1. Further, the performance deficiency involved external events. Consequently a Phase 2 analysis could not be performed and the issue went directly to a Phase 3 analysis. The Senior Reactor Analyst performed a Phase III evaluation in accordance with IMC 609, "Significance Determination Process," and determined that the analysis be performed as a conditional analysis existing for one year. The increase in risk (∆CDF) is solely dependent on the change in the CCDP due to the performance deficiency. The increase in core damage frequency (ΔCDF) for this issue was estimated to be 6.35×10^{-6}. The NRC concluded that the significance of the finding is preliminarily of low to moderate safety significance (White). For the complete analysis, see Enclosure 4 to this inspection report.

This finding had a cross-cutting aspect in the Work Practices component of the Human Performance area because it was directly related to the licensee not ensuring adequate supervisory and management oversight of engineering design work activities associated with a plant design change that adversely affected the operability of both TBBPs. (H.4 (c)).

Enforcement: 10 CFR 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 specified in the license are correctly translated into specifications, drawings, procedures, and instructions. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

UFSAR Section 2.4, "Hydrologic Engineering and WB-DC-40-29, Flood Protection Provisions" and Technical Requirements Manual 3.7.2, "Flood Protection Plan," state that the facility can withstand a Probable Maximum Flood (PMF), and as part of the flood mitigation strategy, the Thermal Barrier Booster Pumps (TBBPs) are assumed to be available.

Temporary Alteration Control Form (TACF) 1-09-0006-070 R0, was developed and completed on December 23, 2009, with the purpose of precluding submergence of the TBBPs during a flooding event. The TBBPs are equipment that could be used to mitigate the effects of a PMF.

Contrary to the above, from 2009 through July 20, 2012, the licensee failed to correctly translate the design basis related to onsite flooding into the instructions for plant design change TACF 1-09-0006-070 R2. Specifically, plant design change TACF 1-09-0006-070 R2 for the TBBP flood protection barrier was inadequate because: (1) the lack of specificity regarding preparation/cleaning of surfaces as recommended by the sealant manufacturer prior to applying the sealant; (2) the failure to perform load calculations for panel deformation; and (3) failure to include provisions in the design and installation to support the temporary panels to resist deflection from hydrostatic pressure/force and potential uplift forces, because the panels were not anchored to the floor. As a result, the TBBP flood barrier would fail during a probable maximum flood event, thereby submerging the TBBPs and rendering the equipment inoperable. Without the TBBPs, the potential greatly increases for a Reactor Coolant Pump (RCP) seal loss of coolant accident. This issue was entered into their corrective action program as PER 619200. This violation is being treated as an AV, consistent with Section 2.3.3 of the NRC Enforcement Policy and is identified as AV 05000390/2012009-04, Failure to Adequately Protect Safety Related Equipment During Flood Mode Preparation.

.5 <u>Failure to Correct Conditions Adverse to Quality Related to Intake Pumping Station (IPS) CKV-040-0604 and IPS 3A and 3B Sump Pumps.</u>

<u>Introduction</u>: The inspectors identified two examples of a non-cited violation (NCV) of the 10 CFR Criteria XVI, “Corrective Action,” for failure to correct conditions adverse to quality for the IPS CKV-040-0604, 0-PUMP-040-003B discharge check valve which resulted in it being non-functional for an extended period of time, and both the IPS 3A and 3B sump pumps, which resulted in the pumps remaining in a degraded condition for an extended period of time.

<u>Description</u>: Licensee records indicated that on March 8, 2008, the 3A sump pump exhibited low flow below the allowable limits of TI-50.021, Intake Pumping Station (IPS) Strainer Room A Sump Pump A Performance Test, and exhibited bubbling from the opposite pump suction 3B, indicating back-leakage past check valve CKV-040-0604, 0-PUMP-040-003B discharge check valve. This resulted in PER 139387 and work order (WO) 08-812124. This PER was closed to previously existing PER 128435, dated August 4, 2007, which was also written for flow-related issues. At the time of the completion of this analysis, WO 08-812124 had not been located and check valve CKV-040-0604 had not been entered into the work control process at that time. PER 128435 was an all-encompassing PER for both train A pumps 3A and 3B.

During the next performance of TI-50.021 on March 1, 2009, the 3A pump yielded zero flow and the operator noted bubbles coming from the opposite sump pump, which again implicated the opposite train check valve 0-CKV-040-0604 as leaking backward past the seat sufficiently to prevent the 3A pump from removing water from the sump it shares with the 3B sump pump. WO 09-812234 was written for this check valve, specifically. This WO was performed, and its associated post maintenance test was signed off as satisfactory on April 15, 2010. TI-50.021 was performed the next day, April 16, 2010, at which time the 3A pump failed on low flow. However, there appeared to be no back-leakage of CKV-040-0604. PER 225913 was written as a result of the test failure which closed to WO 110952174 for check valve 0-CKV-040-0606, 0-PUMP-040-003A discharge check valve for apparently being partially stuck shut. On October 19, 2011, in a situation of high demand due to fire pump strainer leakage, the licensee determined that CKV-040-0604 was stuck open sufficiently to render sump pump 3A incapable of lowering level due to back leakage through CKV-040-0604. As a result, service request (SR) 448624 was initiated and resulted in WO 112833360 which was scheduled to work December 11, 2012. On June 18, 2012, TI-50.021 was performed and failed. As such, no satisfactory testing has been shown to verify the functionality of CKV-040-0604 since April 16, 2010.

Additionally, licensee records indicated that on November 4, 2007, the 3A pump was replaced. Each subsequent test exhibited continual decreasing flow below the allowable limits of TI-50.021, Intake Pumping Station Strainer Room A, Sump Pump A Performance Test, up until the present. On November 2, 2007, the 3B pump was replaced with the power leads reversed leading to reverse rotation and low flow. This was not corrected until January 24, 2008. The pump tested satisfactorily until January 16, 2011 when the scheduled test per TI-50.022, Intake Pumping Station Strainer Room A Sump Pump B Performance Test, was aborted due to a failed breaker disconnect switch which had been in the work planning system since 2009.

On October 19, 2011, in a situation of high demand due to fire pump strainer leakage, the licensee determined that pump 3B would not start in local manual control. Additionally, the 3A pump was pumping, but all flow was being pumped backward through 3B pump. This resulted in an inability of the pumps to remove water from the room. Operator actions were required in this remote structure to stop the level of water rise. As a result, SR 448624 was initiated and resulted in WO 112833360. However, this WO is for 0-CKV-040-0604 and does not mention the start failure of pump 3B. Again on June 17, 2012, TI-50.022 was aborted due to the previously existing deficiency with the breaker disconnect switch. This pump has not been tested since January 24, 2010, and has one intervening known failure to start.

<u>Analysis</u>: The inspectors determined that the 3B sump pump discharge check valve, CKV-040-0604, in the A train of the IPS ERCW strainer room failed to seat on cessation of flow on a number of occasions contrary to the requirements of the augmented in-service testing (IST) program was a performance deficiency.

Additionally, the inspectors determined that the 3A sump pump had repeatedly failed its flow test requirements, and the 3B sump pump had frequently failed its flow or starting requirements contrary to the requirements of the augmented IST program was an additional example of the same performance deficiency.

The performance deficiency was determined to be more than minor because, if left uncorrected would lead to a more significant safety concern. Specifically, internal flooding of the IPS mechanical equipment room housing the train A ERCW strainers could occur over a longer period of time. The inspectors performed a Phase 1 evaluation per IMC 0609, Attachment 4, and determined that the finding was potentially risk significant because it involved the degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors). Consequently a Phase 3 analysis was performed by a Senior Reactor Analyst. The analyst determined the ∆CCDP due to a loss of a single train of ERCW, given a loss of on-site power, potentially would have occurred simultaneously due to the rainfall/flooding event. The analyst then determined the flood frequency that would be required for the finding to be potentially greater-than-green (E-1/year). Given that this was an unrealistically high frequency, the analyst determined that the risk significance of these issues was very low (i.e., $\Delta CDF < 1.0E\text{-}6$).

These findings directly involved the cross-cutting area of Human Performance under the Supervisory and Management Oversight of Work Activities component, in that, the licensee failed to adequately ensure corrective actions were taken to maintain the functionality for extended periods of time. (H.4 (c))

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action,” states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.

Contrary to the above, conditions adverse to quality related to IPS CKV-040-0604, 0-PUMP-040-003B discharge check valve were not corrected, which resulted in it being non-functional for an extended period of time. Additionally, conditions adverse to quality related to both the IPS 3A and 3B sump pumps were not corrected, which resulted in the pumps remaining in a degraded condition for an extended period of time. Because this finding is of very low safety Significance (Green) and has been entered into the corrective action program as PER 597045 and 597047, these issues are being treated as an NCV, consistent with the NRC Enforcement Policy and is identified as NCV 05000390/2012009-05, Failure to Correct Conditions Adverse to Quality Related to IPS CKV-040-0604 and IPS 3A and 3B Sump Pumps.

4OA6 Meetings, including Exit

On February 21, 2013, the resident inspectors presented the inspection results to Mr. Tim Cleary, Site Vice President, and other members of the licensee staff. Also in attendance was Scott Shaeffer, Chief, Reactor Projects Branch 6. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.

Contrary to the above, conditions adverse to quality related to the main control room chilled water circulating pumps A-A and B-B and shutdown board room chilled water circulating pumps A-A and B-B were not corrected which resulted in the chillers being inoperable and reportable. The inspectors performed a Phase 1 evaluation per IMC 0609, Attachment 4, and determined that the finding was potentially risk significant because it involved the degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors). Consequently, a Phase 3 analysis was performed by a Senior Reactor Analyst. The analyst estimated the frequency of the PMP event, adjusted for the time of year when room cooling would be necessary and multiplied by a conservative value for ΔCCDP (0.1) representing the likelihood for core damage due to alternate shutdown from outside the control room. The analyst determined that the risk significance of the issue was very low (i.e., ΔCDF < 1.0E-6). Because this finding is of very low safety significance (Green) and has been entered into the corrective action program as PER 641937, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Bankes, Interim Chemistry/Environmental Manager
T. Carter, Interim Site Engineering Director
T. Cleary, Interim Site Vice President
T. Detchemende, Emergency Preparedness Manager
R. Dittmer, Operations Superintendent
W. Francis, Interim Maintenance Manager
D. Gronek, Plant Manager
D. Guinn, Licensing Manager
E. Higgins, Civil Design Manager
W. Hooks, Radiation Protection Manager
D. Hughes, Training Supervisor
B. Hunt, Operations Support Superintendent
D. Jacques, Security Manager
R. Kirkpatrick, Design Engineering Manager
W. Prevatt, Operations Manager
A. Scales, Work Control Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000390/2012009-01	AV	Inadequate Abnormal Condition Procedure for Flood Mitigation Strategy Prior to Installation of HESCO Barriers (Section 1R01.1)
05000390/2012009-02	AV	Failure to Report Unanalyzed Condition Related to External Flooding (Section 1R01.2)
05000390/2012009-03	AV	Failure to Maintain an Adequate Abnormal Condition Procedure to Implement the Flood Mitigation Strategy (Section 1R01.3)
05000390/2012009-04	AV	Failure to Adequately Protect Safety-Related Equipment During Flood Mode Preparation (Section 1R01.4)

Opened and Closed

05000390/2012009-05	NCV	Failure to Correct Conditions Adverse to Quality Related to IPS CKV-040-0604 and IPS 3A and 3B Sump Pumps (Section 1R01.5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: External Flood Protection Inspection
Technical Requirements Manual 3.7.2, Flood Protection Plan
Watts Bar UFSAR Section 2.4, Hydraulic Engineering

Design Basis Documents (Functional System Descriptions)
WB-DC-40-39, Flood Protection Provisions, Rev. 11
WB-DC-40-69, Design Criteria for Electrical and Mechanical Penetration Seal Assemblies for Category I Structures
TACF 1-09-0006-070, Rev. 0, 2, 3, 4, 5

Calculations
CDQ000020080054, Rev. 0, 1, 2 and 3 PMF Determination for Tennessee River Watershed
CDQ000020080009, Rev. 2, Initial Dam Rating Curve Fort Loudoun
CDQ000020080080, Rev. 2, Flood Levels at WBN and SQN from Seismic Dam Failures

Procedures
AOI-7.1, Maximum Probable Flood
AOI-7.10, Flood Mode Electrical Systems Alignment
MI-17.021, Installation of Spool Pieces between ERCW and Component Cooling Systems

Corrective Action Documents (PERs)
138749, attachment TVA Hydrology Model Issue Identification and Assessment, dated 5/14/2010, pages 12 and 13
177501 associated functional evaluation Rev. 1

138749	179244	499217
177492	202572	519131
177669	202693	573093
177822	202777	580109
178649	202723	582541
179001	202622	582543
179338	211722	

LIST OF ACRONYMS

AFW	auxiliary feedwater
AOI	Abnormal Operating Instruction
AV	Apparent Violation
BLN	Bellefonte Nuclear Plant
CCS	component cooling system
CFR	*Code of Federal Regulations*
COLA	Combined License Application
DBF	design basis flood
ERCW	essential raw cooling water
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
IPS	intake pumping station
IST	in-service testing
LOCA	loss of coolant accident
NCV	Non-Cited Violation
NGDC	New Generation and Design Construction
NOV	Notice of Violation
NPG	Nuclear Power Group
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Report
PMF	probable maximum flood
PMP	probable maximum precipitation
QA	Quality Assurance
RCP	reactor coolant pump
RCW	raw cooling water
Rev.	revision
RHR	residual heat removal
RTV	room temperature vulcanizing
SDP	Significance Determination Process
SFP	spent fuel pool
SR	Service Request
TACF	Temporary Alteration Control Form
TBBP	thermal barrier booster pump
TRM	Technical Requirements Manual
TS	Technical Specification
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
WO	Work Order