UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

November 17, 2011

EA-11-018
EA-11-252

Mr. Joseph W. Shea
(Acting) Vice President
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT – NRC INSPECTION PROCEDURE 95003 SUPPLEMENTAL INSPECTION REPORT 05000259/2011011, 05000260/2011011, AND 05000296/2011011 (PART 1)

Dear Mr. Shea:

On September 23, 2011, the U.S. Nuclear Regulatory Agency (NRC) completed Part 1 of a supplemental inspection pursuant to Inspection Procedure 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," at your Browns Ferry Nuclear Station, Unit 1. The enclosed inspection report documents the inspection results, which were discussed at the exit meeting on October 3, and November 3, 2011, with Mr. Preston Swafford, Mr. Tim Cleary, Mr. Keith Polson and other members of the TVA staff.

The NRC's Reactor Oversight Process (ROP) collects information to enable the agency to arrive at objective conclusions about a licensee's safety performance. The assessment information is used to determine the appropriate agency response. The NRC's Action Matrix, found in Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," delineates expected NRC and licensee actions based on the inputs to the assessment process. Agency action beyond the baseline inspection program will normally occur only if assessment input thresholds are exceeded. The Action Matrix identifies the range of NRC and licensee actions and the appropriate level of communication for varying levels of licensee performance. The Action Matrix describes a graded approach in addressing performance issues.

As required by the NRC ROP Action Matrix, this supplemental inspection was performed because one finding of red safety significance was identified which placed Browns Ferry Unit 1 in the Multiple/Repetitive Degraded Cornerstone Column in the fourth quarter of 2010. The issue, which degraded the Mitigating Systems Cornerstone, was a Red finding for the Residual Heat Removal Subsystem being inoperable for greater than the Technical Specification allowed outage time. This issue was documented in NRC Inspection Report 05000259/2011008, dated May 9, 2011 (ML111290482). The objectives of this inspection were to provide the NRC with information regarding Browns Ferry's: (1) maintenance and testing program related to inservice testing (IST); (2) maintenance and testing program related to motor operated valves (MOVs);

and (3) the corrective action program (CAP), to include the immediate corrective actions taken to address the red finding. This inspection also evaluated the broader extent of condition aspects of the testing programs used at the station to comply with technical specifications and other regulatory requirements. Additionally, this inspection was conducted to provide the NRC additional information to be used in deciding whether the continued operation of the facility is acceptable and whether additional regulatory actions are necessary to prevent declining plant performance. The inspection consisted of examination of activities conducted under your license as they related to safety, compliance with the Commission's rules and regulations, and the conditions of your operating license. The NRC concluded that the evaluated Browns Ferry Programs generally meet the requirements of the NRC's rules and regulations. The inspection identified several issues involving programmatic requirements and implementation of those programs.

Based on the results of this inspection, one apparent violation involving 10 CFR 50.9, "Completeness and Accuracy of Information," was identified and is being considered for escalated enforcement action in accordance with the NRC's Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at http://pbadupws.nrc.gov/docs/ML0934/ML093480037.pdf. Specifically, by letter dated January 6, 1997, TVA provided its response to a prior NRC request for reevaluation of the safety functions of certain Motor Operated Valves (MOVs) to be included in the BFN Unit 2 and 3 Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" testing program. TVA's letter of January 6, 1997, was in regard to whether valves FCV-74-52 and FCV-74-66 had a redundant safety function to close to allow operation of the suppression pool cooling mode of the RHR System, and stated that "Closure of valves FCV-74-52 and FCV-74-66 is not required by plant procedures to operate the RHR system in the suppression pool cooling mode. Therefore, these valves have no "redundant" safety function and will not be included in the GL-89-10 program." The NRC concluded that this information was inaccurate because valves FCV-74-52 and FCV-74-66 do have a safety function to close to operate the RHR system in the suppression pool cooling mode, as described in Emergency Operating Instruction (EOI) Appendix-17A, RHR System Operation Suppression Pool Cooling.

Additionally, TVA's letter of May 5, 2004 provided its updated response to NRC GL 89-10 for BFN Unit 1. TVA's updated response included a listing of 18 valves, which included valves FCV-74-52 and FCV-74-66, "that are not in the GL 89-10 program, since the valves are normally in their safety position." TVA's May 5, 2004 letter also referenced its previous January 6, 1997 letter regarding similar valves on Units 2 and 3 (including FCV-74-52 and FCV-74-66) that were not in the GL 89-10 program, since the valves are normally in their safety position. The NRC concluded that the information provided in TVA's May 5, 2004 letter was incomplete, in that it did not discuss or acknowledge that Unit 1 valves FCV-74-52 and FCV-74-66 in fact have a redundant safety function to close to allow operation of the suppression pool cooling mode of the RHR System, as described in EOI Appendix-17A.

The above information was material to the NRC because it was used, in part, as the basis for determining that valves FCV-74-52 and FCV-74-66 for Units 1, 2, and 3 did not meet the conditions necessary to require incorporation into BFN's GL 89-10 MOV monitoring program. The NRC determined that had these valves been included in the licensee's MOV monitoring program, the identification of the previously failed Unit 1 FCV-74-66 may have been identified earlier by the comprehensive testing that would have been implemented by the monitoring program. As required by 10 CFR 50.9(b), TVA provided written notification to the NRC by letter dated October 20, 2011, acknowledging the inaccuracy of its January 6, 1997 letter.

Before the NRC makes its enforcement decision, we are providing you an opportunity to either: (1) respond to the apparent violation addressed in this inspection report within 30 days of the date of this letter; or (2) request a Pre-Decisional Enforcement Conference. If a PEC is held, it will be open for public observation and the NRC will issue a press release to announce the time and date of the conference. If you decide to participate in a PEC, please contact Mr. Eugene Guthrie at (404) 997-4662 within 10 days of receipt of this letter to notify the NRC of your intended response. A PEC should be held within 30 days of the date of this letter.

If you choose to provide a written response, it should be clearly marked as a "Response to an Apparent Violation in Inspection Report 05000259/2011011, 05000260/2011011, 05000296/2011011; EA-11-252" and should include: (1) the reason for the apparent violation, or, if contested, the basis for disputing the apparent violation; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addressed 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 or schedule a PEC.

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 that 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 a violation occurred, information to determine the significance of a violation, information related to the identification of a violation, and information related to any corrective actions taken or planned to be taken.

Additionally, based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance and they were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with the NRC Enforcement Policy. If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Browns Ferry Nuclear Plant. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Browns Ferry Nuclear Station.

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 any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Richard P. Croteau
Division Director
Division of Reactor Projects

Docket Nos.: 50-259, 50-260, 50-296
License Nos.: DPR-33, DPR-52, DPR-68

Enclosure: Inspection Report 05000259/2011011, 05000260/2011011, and 05000296/2011011
w/Attachment: Supplemental Information

cc w/encl.: (See page 5)

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cc w/encl.: (See page 5)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: ☐ Yes ACCESSION NUMBER: ML113210602 ☐ SUNSI REVIEW COMPLETE

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NAME	LLake	JHuang	MFarnan	JNadal	ROrlikowski	MAshley	LCasey
DATE	11/14/2011	11/16/2011	11/16/2011	11/16/2011	11/16/2011	11/16/2011	11/16/2011
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
OFFICE	OGC	RII/OE	RII/DRP	RII/DRP	RII/DRP		
SIGNATURE	Via Email	Via email	Via EMail	Via email	RPC /RA/		
NAME	CHair	SSparks	CKontz	EGuthrie	RCroteau		
DATE	11/16/2011	11/16/2011	11/16/2011	11/16/2011	11/17/2011		
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Letter to Joseph W. Shea from Richard P. Croteau dated November 17, 2011

SUBJECT: BROWNS FERRY NUCLEAR PLANT – NRC INSPECTION PROCEDURE 95003 SUPPLEMENTAL INSPECTION REPORT 05000259/2011011, 05000260/2011011, AND 05000296/2011011 (PART 1)

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U.S. Nuclear Regulatory Commission

Region II

Docket Nos.:	50-259, 50-260, and 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2011011, 05000260/2011011, 05000296/2011011
Licensee:	Tennessee Valley Authority
Facility:	Browns Ferry Nuclear Station Units 1, 2, and 3
Location:	Athens, AL 35611
Dates:	September 12, 2011, through September 23, 2011
Inspectors:	R. Orlikowski, Project Engineer, Team Leader L. Lake, Senior Reactor Inspector J. Huang, Senior Mechanical Engineer M. Farnan, Mechanical Engineer J. Nadel, Resident Inspector
Approved By:	Richard P. Croteau Division Director Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2011011, 05000260/2011011, 05000296/2011011; Browns Ferry Nuclear Plant, Units 1, 2, and 3; 09/12/2011- 09/23/2011; Browns Ferry Nuclear Station Units 1, 2, and 3; Supplemental Inspection – Inspection Procedure (IP) 95003.

This supplemental inspection was conducted by a Project Engineer, a Senior Reactor Inspector, a Senior Mechanical Engineer, a Mechanical Engineer, and a Resident Inspector. Three findings and one Apparent Violation were identified. The NRC's program for overseeing the safe operations of commercial nuclear reactor power reactors is described in the NUREG-1649, "Reactor Oversight Process."

The NRC staff performed Part 1 of this supplemental inspection in accordance with IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," to evaluate Browns Ferry Nuclear Station's maintenance and testing programs. The results of this inspection, when combined with the results from Parts 2 and 3 of the Browns Ferry Inspection Procedure (IP) 95003 inspection, will allow the NRC to determine the breadth and depth of safety, organizational, and programmatic issues at Browns Ferry. This Part 1 inspection focused specifically on maintenance and testing programs related to inservice testing (IST), the motor operated valve (MOV) testing, and the corrective action program (CAP). The team additionally inspected broader extent of condition aspects of the testing programs used at the station to comply with technical specifications and other regulatory requirements. Additionally, this inspection was intended to provide the NRC additional information to be used in deciding whether the continued operation of the facility is acceptable and whether additional regulatory actions are necessary to prevent declining plant performance.

A. NRC-Identified & Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a NCV of very low safety significance for the licensee's failure to implement a procedure required by Technical Specification (TS) 5.5.6, Inservice Testing Program. Specifically, inspectors determined that TVA did not adequately implement 0-TI-362, "Inservice Testing of Pumps and Valves.", which required that a Service Request (SR) be documented in the CAP when pumps are found to be in the Alert Range during inservice testing.

  The inspectors determined that the failure to implement procedure 0-TI-362 constituted a performance deficiency. This performance deficiency was determined to be more than minor in accordance with IMC 0612 Appendix B, "Issue Screening" because if left uncorrected, the failure to enter degraded conditions in the CAP has the potential to lead to a more serious safety concern. The inspectors screened this finding in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green). The cause of this finding was directly related to the cross-cutting area of problem identification and resolution, the component of the corrective action program and the aspect of issue identification; because the licensee failed to implement the corrective action program. [P.1(a)] (Section 4OA4.2.d(1))

- Green. The inspectors identified an NCV of very low safety significance involving the licensee's failure to implement a procedure as required by 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The inspectors determined that TVA failed to implement the requirements of procedure NEPD-22, "Functional Evaluations," when the licensee's failed to verify the technical veracity and perform an adequate review, of multiple functional evaluations written to support operability of MOVs which experienced overthrust conditions.

  The finding was determined to be more than minor because if left uncorrected, could become a more significant safety concern. Specifically, the review of the functional evaluation that is performed by the engineering supervisor is a critical part of the functional evaluation process to second check the quality of work that will ultimately be provided to the Senior Reactor Operator (SRO). The SRO will use this evaluation to aid in determining equipment operability. Therefore, the failure to thoroughly review Functional Evaluations and identify discrepancies could lead to incorrect information being used to determine equipment operability. The inspectors determined this finding was associated with the Mitigating Systems Cornerstone and characterized in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone. The finding screened as having very low safety significance (Green) because the finding was a design or qualification deficiency confirmed not to result in a loss of operability at this time.

  This finding has a cross-cutting aspect in the area of human performance, decision making, because the licensee did not make safety-significant or risk-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. Specifically, on multiple occasions, the engineering supervisor responsible for reviewing an engineering evaluation that was to be provided to the SRO for a degraded equipment condition failed to implement his/her role and authority in reviewing and approving functional evaluations. The Supervisors failed to provide the interdisciplinary review during the decision making process that should have identified that a Technical Update was not applicable to safety related valve actuators. [H.1(a)] (Section 4OA4.3.l(2))

- Green. The inspectors identified a finding of very low safety significance and associated non-cited violation of 10 CFR Part 50.55a(b)(3)(ii), for the licensee's failure to adequately reestablish the design basis capability of multiple Motor Operated Valves (MOVs) after internal modifications were performed to the valves.

  The NRC inspectors determined that the methodology described in the TVA documentation for justifying the design-basis capability of MOVs and the specific justification prepared by TVA to reestablish the design basis capability of MOVs after undergoing internal modifications did not satisfy 10 CFR 50.55a(b)(3)(ii), and was a performance deficiency. Further, this was determined to be a programmatic issue because there were at least 12 examples of other modified MOVs where the licensee implemented its methodology that did not provide an adequate justification for the design basis capability of those MOVs that would satisfy 10 CFR 50.55a(b)(3)(ii). The NRC staff determined the finding to be more than minor because the program deficiency, if left uncorrected, could become a more significant safety concern. Specifically, by establishing a design basis MOV valve factor that TVA considered to be conservative using data from two tested valves obtained from the JOG MOV Performance Verification program without demonstrating its applicability to the Browns Ferry valves, BFN personnel might not realize that the established valve factor is the minimum value that must be used to set up MOV actuator operating parameters. The inspectors concluded this finding was associated with the Mitigating Systems Cornerstone. The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone. The finding screened as having very low safety significance (Green) because the finding was a design or qualification deficiency confirmed not to result in a loss of operability at this time. This finding had a cross-cutting aspect in the area of human performance, decision making, because the licensee failed to use conservative assumptions in decision making. Specifically, the licensee made modifications to multiple safety related MOVs and then reestablished their design basis capability using methods that were inconsistent with industry and NRC guidance. [H.1(b)] (Section 4OA4.3.l(1))

Other: Enforcement

- To Be Determined (TBD). An NRC-identified apparent violation of 10 CFR 50.9(a) requirements was identified when it was determined that the licensee provided information that was not complete and accurate in the letter dated January 6, 1997, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 89-10, Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance, NRC Inspector Follow-up Item (IFI) 50-260, 296/95-19-01, Response to Request for Reevaluation Regarding Reduced Scope of MOVs." Additionally, TVA provided incomplete and inaccurate information to the NRC in a letter from T. E. Abney, "Browns Ferry Nuclear Plant (BFN) Unit 1 – Generic Letter 89-10 and Supplements 1 to 7, Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance," dated May 5, 2004. This was an apparent violation of 10 CFR 50.9, "Completeness and Accuracy of Information."

  The inspectors determined that the failure to provide complete and accurate information to the NRC was contrary to the requirements of 10 CFR 50.9, and was an apparent violation. Because violations of 10 CFR 50.9 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process. The regulatory significance was important because this information was material to the NRC because it was used, in part, as the basis for determining that valves FCV-74-52 and FCV-74-66 did not meet the conditions necessary that would require them to be in Browns Ferry's

GL 89-10 MOV monitoring program. The issue was preliminarily determined to be an apparent violation of 10 CFR 50.9. (Section 4OA4.5.b(1))

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA4 Supplemental Inspection (95003)

.1 Inspection Scope

This inspection was conducted in accordance with Inspection Procedure (IP) 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," to assess the licensee's evaluation of one Red Finding. The inspection objectives were to provide the NRC with information regarding Browns Ferry's:

- Corrective action program, to include the immediate corrective actions taken to address the red finding (IP 95001);
- Maintenance and testing program related to the motor operated valve program (IP 62708); and
- Maintenance and testing program related to the IST program (IP 73756).

Browns Ferry Nuclear Plant entered the Multiple/Repetitive Degraded column of NRC's Action Matrix in the fourth quarter of 2010. The issue, which degraded the Mitigating Systems Cornerstone, was a finding of high safety significance (Red), for the Residual Heat Removal Subsystem being inoperable for greater than the Technical Specification allowed outage time. These issues were documented in NRC Inspection Report 05000259/2011008, dated May 9, 2011. (ML111290482).

.2 IP 95001 Requirements

a. Problem Identification

(1) *Determine whether the evaluation identified who (i.e., licensee, self revealing, or NRC), and under what conditions the issue was identified.*

The inspector determined that the root cause evaluation adequately identified who and under what conditions the issue was identified. Specifically, the executive summary of the revision 1 root cause describes the events that led to the identification that the disc was separated from the stem of 1-FCV-74-66. It is clear that the issue self-revealed when no flow was seen after the attempt to place Residual Heat Removal (RHR) loop 2 in service on October 23, 2010.

(2) *Determine whether the evaluation documented how long the issue existed, and whether there were any prior opportunities for identification.*

The inspector determined that the root cause evaluation adequately identified how long the issue existed and whether there were any prior opportunities for identification.

The revision 1 root cause evaluation concluded that the disc to stem separation occurred sometime before November of 2008. Several prior opportunities for identification were mentioned throughout the revision 1 root cause.

The inspectors found the licensee's assessment in this area to be adequate and considered the valve inoperable and unable to perform its safety function since at least March 13, 2009.

(3) Determine whether the licensee's root cause evaluation documented the plant specific risk consequences and compliance concerns associated with the issue.

Inspectors noted, and the licensee acknowledged, that the revision 1 root cause was narrow in both the scope of the identified cause and in the corresponding corrective actions. The NRC has determined that there are programmatic deficiencies in both the licensee's maintenance and testing programs and their CAP. The revision 1 of the root cause did not evaluate these deficiencies as a possible cause or create corrective actions to address them. The licensee informed the inspectors they were working on changes to the root cause during the inspection in order to address these concerns. The final root cause will be evaluated by the NRC in a later phase of the 95003 inspection.

b. Root Cause, Extent of Condition, and Extent of Cause Evaluation

In evaluating the root cause, the inspectors noted, and the licensee acknowledged, that the revision 1 of the root cause was narrow in both the scope of the identified cause and in the corresponding corrective actions. The NRC has determined that there are programmatic deficiencies in both the licensee's maintenance and testing programs, and their CAP. The revision 1 of the root cause did not evaluate these deficiencies as a possible cause or create corrective actions to address them. The licensee informed the inspectors they were working on changes to the root cause during the inspection in order to address these concerns. The final root cause will be evaluated by the NRC in a later phase of the 95003 inspection for the following:

1) *Determine whether the licensee's root cause evaluation applied systematic methods in evaluating the issue in order to identify root causes and contributing causes.*
2) *Determine whether the licensee's root cause evaluation was conducted to a level of detail commensurate with the significance of the problem.*
3) *Determine whether the licensee's root cause evaluation included consideration of prior occurrences of the problem and knowledge of prior operating experience.*
4) *Determine whether the licensee's root cause evaluation addressed extent of condition and extent of cause of the problem.*
5) *Determine whether the licensee's root cause evaluation, extent of condition, and extent of cause appropriately considered the safety culture components as described in IMC 0305.*

c. Corrective Actions

(1) *Determine whether the licensee specified appropriate corrective actions for each root/contributing cause or that the licensee evaluated why no actions were necessary.*

The inspector determined that the licensee specified appropriate corrective actions to correct the immediate safety concerns associated with 1-FCV-74-66 and the other valves affected by the extent of condition.

The revision 1of the root cause evaluation concluded that the valve failed because the opening thrust exceeded the strength of the threaded connection between the disc skirt and disc due to a manufacturing defect in which the threads were undersized. Four main corrective actions to prevent recurrence were identified:

1. Verify the design configuration of 1-FCV-074-0052 and 2, 3-FCV-074-066/-052 valves and rework as required to ensure the valves have the correct design configuration.

2. Rework 1-FCV-074-066 valve using revised drawing 0-A-12337-M-1E. Re-assembly of the valve with the skirt keyed to the stem, and the skirt welded to the disc will return the valve to the correct design configuration.

3. Evaluate the root cause findings “Undersized Disc Skirt Threads” as a manufacture’s defect in accordance with 10 CFR Part 21.

4. Report the findings of the 10CFR Part 21 to BFN Licensing.

In addition, the licensee performed a number of immediate corrective actions, including troubleshooting and rework of the failed 1-FCV-74-66 valve, drawing corrections, and functionality evaluations for other potentially affected valves. Borescope and ultrasonic testing inspections were also performed on the other potentially affected valves.

No other root or contributing causes were identified in the revision 1 of the root cause evaluation.

The inspectors concluded that the corrective actions delineated above were adequate to address the immediate safety concerns associated with 1-FCV-74-66 and the other valves affected by the extent of condition. However, inspectors identified that even though the corrective actions to prevent recurrence were completed, the licensee failed to enter them into the CAP. This issue was determined to be minor because the corrective actions were completed. The licensee wrote Problem Evaluation Report (PER) 433927 to address this issue. In evaluating the root cause, the inspectors noted, and the licensee acknowledged, that the revision 1 of the root cause was narrow in both the scope of the identified cause and in the corresponding corrective actions. The NRC has determined that there are programmatic deficiencies in both the licensee’s maintenance and testing programs, and their CAP. The revision 1 of the root cause did not evaluate these deficiencies as a possible cause or create corrective actions to address them. The licensee informed the inspectors they were working on changes to the root cause during the inspection in order to address these concerns. The final root cause will be evaluated by the NRC in a later phase of the 95003 inspection.

(2) *Determine whether the licensee prioritized the corrective actions with consideration of the risk significance and regulatory compliance.*

The licensee's corrective actions were listed in the revision 1 root cause with an associated due date for each action. No detail was provided to indicate what, if any, risk significance or regulatory compliance concerns were used in determining the prioritization of the listed corrective actions.

The inspectors did conclude however, that both the immediate corrective actions and the corrective actions to prevent recurrence were carried out at the first available opportunity and in a manner commensurate with their safety significance. In evaluating the root cause, the inspectors noted, and the licensee acknowledged, that the revision 1 of the root cause was narrow in both the scope of the identified cause and in the corresponding corrective actions. The NRC has determined that there are programmatic deficiencies in both the licensee's maintenance and testing programs, and their CAP. The revision 1 of the root cause did not evaluate these deficiencies as a possible cause or create corrective actions to address them. The licensee informed the inspectors they were working on changes to the root cause during the inspection in order to address these concerns. The final root cause will be evaluated by the NRC in a later phase of the 95003 inspection.

(3) *Determine whether the licensee established a schedule for implementing and completing the corrective actions.*

The inspector determined that the licensee adequately established a schedule for implementing and completing the corrective actions.

The licensee set scheduled due dates for each identified corrective action in the revision 1 root cause.

The inspectors concluded that the licensee has adhered to their scheduled dates.

(4) *Determine whether the licensee developed quantitative or qualitative measures of success for determining effectiveness of the corrective actions to prevent recurrence.*

The licensee has determined that corrective actions to prevent recurrence will be considered effective with no recurring issues and satisfactory valve operability as determined through periodic technical specification required surveillance testing. Additionally, the licensee will consider satisfactory valve performance through MOV program trending.

In evaluating the root cause, the inspectors noted, and the licensee acknowledged, that the revision 1 of the root cause was narrow in both the scope of the identified cause and in the corresponding corrective actions. The NRC has determined that there are programmatic deficiencies in both the licensee's maintenance and testing programs, and their CAP. The revision 1 of the root cause did not evaluate these deficiencies as a possible cause or create corrective actions to address them. The licensee informed the inspectors they were working on changes to the root cause during the inspection in order to address these concerns. The final root cause will be evaluated by the NRC in a later phase of the 95003 inspection.

d. Findings

.1 Failure to Implement Requirements of the Inservice Testing Program

Introduction: The inspectors identified an NCV of very low safety significance (Green) for the licensee's failure to implement a procedure required by TS 5.5.6, Inservice Testing Program. Specifically, inspectors determined that TVA did not adequately implement 0-TI-362, "Inservice Testing of Pumps and Valves", which required that a Service Request (SR) be documented in the CAP when pumps are found to be in the Alert Range during IST.

Description: On September 20, the inspectors identified that the licensee had a population of safety related pumps in the alert range for either vibrations or differential pressure (flow) and as a result, were being tested at double frequency in accordance with the requirements of the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM code). The affected pumps included Residual Heat Removal Service Water (RHRSW) pumps A1, A2, B1, D1, D2, Unit 1 RHR pump D1, and the Unit 1 HPSI pump. TVA procedure 0-TI-362, "Inservice Testing of Pumps and Valves" required "Test results which place a pump as operating in the Alert Range or Required Action Range shall be documented in the site's CAP and evaluated accordingly." Upon further inspection, the inspectors noted that the licensee was only writing corrective action documents when a pump would test in the Required Action Range, causing the surveillance test to fail and the pump to be declared inoperable. The inspectors identified six instances where pumps were in the Alert Range for long periods of time without any documentation in the CAP. Some were not documented in the CAP until conditions degraded to the point where the pump was declared inoperable or it catastrophically failed. Pumps that test in the Alert Range, while not inoperable, may be indicative of a degrading or deficient condition. Documentation of such issues in the CAP is essential to ensure that appropriate corrective actions are taken to resolve the issue.

Analysis: The inspectors determined that the failure to implement procedure 0-TI-362 constituted a performance deficiency. This performance deficiency was determined to be more than minor in accordance with IMC 0612 Appendix B, "Issue Screening," because if left uncorrected the failure to enter degraded conditions in the CAP has the potential to lead to a more serious safety concern. The inspectors screened this finding in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green) since it was determined not to be potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of this finding was directly related to the cross-cutting area of problem identification and resolution, the component of the CAP and the aspect of issue identification; because the licensee failed to implement the CAP. [P.1(a)]

Enforcement: Brown's Ferry TS section 5.5, "Programs and Manuals," required that the listed programs be established, implemented, and maintained. Specifically, specification 5.5.6 required the establishment and implementation of an IST Program. 0-TI-362, "Inservice Testing of Pumps and Valves" was an implementing procedure of the IST program. Contrary to the above, the licensee failed to implement procedure 0-TI-362, "Inservice Testing of Pumps and Valves" on at least six occasions prior to September 20, 2011, when pumps where tested in the Alert Range but no SRs were initiated in the CAP. Because this violation was of very low safety significance (Green) and was entered into the licensee's CAP as SRs: 436076, 436077, 436078, 436079, 436080, 436081, 436083, and 436073, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy (NCV 05000259, 260, 296/2011011-01: Failure to Implement Requirements of the Inservice Testing Program).

.2 IP 62708, Motor Operated Valve Capability

a. MOV Selection: Select a sample of risk-significant MOVs from more than one safety-related system. The selection of MOVs should also include consideration of various valve sizes, types, and manufacturers. The sample size should be appropriate for the scope of the inspection

Review of the failed MOV's 1-FCV-074-0052 and 1-FCV-074-0066 revealed that these valves had a safety function to close and should have been considered active and placed into the MOV 89-10 program. A comparison of the IST program found discrepancies from the MOV program in that certain valves were classified as passive in the MOV program but classified active in the IST program. The licensee performed an internal review and concluded that eight more MOV's should be changed from passive to active and included in the MOV 89-10 program scope. It was also noted that balance of plant MOV's did not follow any type of 89-10 scoping for PM activity. However, BFN Unit 1 had adopted collecting MCC motor current data for analysis and trending. The licensee stated that this practice will be adopted by Units 2 and 3.

The inspectors found that the licensee was in the process of implementing the final program for meeting GL 96-05. The licensee has classified all their MOV's and had implemented many modifications to improve valve functionality and margin.

b. MOV Program Scope: Review MOV program scope changes since the completion of the GL 89-10 program reviews to determine that the appropriate safety-related MOVs are included in the program

A review of MOV program scope since completion of GL 89-10 was completed. Historical review noted that the licensee had a major change in program scope in 1996. The licensee claimed that several MOV's did not have an active safety function and reclassified them as System Operational Enhancement (SOE) and would be exempt from GL 89-10 provisions. The NRC staff reviewed the licensee's analysis and responded that the staff did not agree. The licensee re-analyzed the MOV population and placed a number of valves back into the program. However, 74-66 and 74-52 valves were still considered out of scope at this time. This precluded the valves from undergoing more extensive testing to assure functional capabilities of the valves.

An NRC Independent review panel concluded that the 74-66 and 74-52 valves had a safety function to close and should be included in the MOV program (See Section 4OA4.5.b(1) for further discussion on this issue). The licensee performed an independent review and discovered 8 more MOV's that were incorrectly classified as passive and should have been classified as active and included in the MOV program. These valves were: 74-01, 74-02, 74-12, 74-13, 74-24, 74-25, 74-35, and 74-36.

c. Design Calculations: Review design documents and calculations for: MOV functional requirements under normal, abnormal, and accident conditions; motor and actuator sizing; methods for selecting, setting, and adjusting MOV switch settings; and modifications to the system or valves that could affect the MOV's capability in the as-modified configuration

A review of the methodology for MOV design calculations was completed. The inspectors determined that the licensee incorporated all the necessary design calculations and was in the process of incorporating Electric Power Research Institute (EPRI) Predictive Performance Model (PPM) calculations. PPM is a conservative alternative calculation designed for those valves that cannot be dynamically tested.

d. Testing: Review test documents for adequacy of test procedures, test equipment, training of test personnel, acceptance criteria, and test results. If the inspection schedule permits, observe actual testing of MOVs

Test procedures, equipment, training, acceptance criteria, and test results were found to be adequate. Observation of a field test of a motor-operated valve that was experiencing leakage was performed. The field test was designed to gather data to insure that the valve and actuator were set up as left from the previous static test. The maintenance personnel configured the test equipment and obtained diagnostic data. A field engineer analyzed the data and confirmed that the valve and actuator was set as left from the previous test.

One observation noted that the maintenance personnel performing the MOV test were capable but apprehensive during the test activity. The inspectors found that online testing was not routinely performed, rather.the majority of the MOV testing was completed during refueling outages. The licensee stated that contractors were hired to do the bulk of the MOV testing during outages. The inspectors found that the MOV testing staff did not have a high level of proficiency with online MOV testing and analysis.

e. <u>MOV Trending: Review MOV trend reports, failure analyses, corrective actions, nonconformance reports, or other plant documents that may indicate that an MOV is not properly sized, has improper switch settings, or is not properly maintained</u>

A review of MOV trending was completed. The inspectors determined that the licensee was trending the minimum industry parameters for a typical healthy MOV program.

One observation was that the licensee's MOV program incorporated an assumed 0.15 stem-to-stem nut coefficient of friction (COF) into all MOV design calculations. The basis was captured in Mechanical Design Standard DS-M18.2.21, "Motor Operated Valve Thrust and Torque Calculation." DS-M18.2.21 credited excellent maintenance and preventive maintenance practices to maintain the 0.15 COF assumed friction factor in the design calculations. However, the MOV trending program didn't specifically look at the stem-to-stem nut COF and trend its value based on periodic static testing. The COF value was combined with the overall margin and trended. This type of trend didn't allow for an easy method of periodically verifying the mechanical design standard assumption of 0.15 COF.

Another inspector observation identified several PERS that acknowledged that MOV trend reports did not meet the reporting standard of 90 days after completion of the refueling outage. The inspectors found that the licensee's response to many of the PERS was that the MOV program engineer was burdened with implementing the final stages of GL 96-05 and were not able to complete the trend report. The inspectors were informed that the licensee recently hired an experienced MOV engineer.

f. <u>Preventive Maintenance: Review MOV preventive maintenance to determine whether it is appropriate for the frequency of operation, working environment, and operational experience</u>

Preventive maintenance tasks were reviewed and found to be appropriate for type and interval.

g. <u>Corrective Actions: Determine whether the licensee is periodically reviewing data on MOV failures and the effectiveness of the corrective actions</u>

The inspectors chose a sample of MOV issues entered into the CAP to determine whether the licensee was evaluating the effectiveness of correct actions. Additionally, the inspectors reviewed a sample of MOV maintenance logs and work orders to determine whether the licensee was reviewing data on MOV failures.

The inspectors found that the licensee did review data on MOV failures, however the inspectors noted that the reviews occasionally lacked rigor. One specific example was related to PER 15777 which was written to document "MOV PERs related to Unit 1 restart workmanship." This PER listed 25 PERs written to document MOV issues that arose during the Unit 1 restart, but the purpose of the PER was to pass on Operating Experience (OE) for Watts Bar restart. Review of the "Browns Ferry NP Trending Report for U1C7" did not identify any adverse trends. This trending report included information about 4 valves that were over thrust, 2 valves that had wedges installed backwards, an incorrectly installed motor pinion gear, excessive grease found in the spring pack of 6 operators, and 6 PERs for Local Leak Rate Test failures due to installation and setting limits on new double disc gates. A more thorough evaluation of the information contained in PER 15777 could have provided valuable insights for the licensee to improve their practices related to MOVs.

h. Post-Maintenance Testing: Review a sample of MOV maintenance packages and verify that the post-maintenance tests and results demonstrate that the MOVs are capable of performing their design functions

A random selection of motor-operated valve diagnostic test data was reviewed. The diagnostic test traces were evaluated for proper valve set up, completeness, accuracy, and quality of analysis. No concerns were identified. The data reviewed demonstrated that MOVs were capable of performing their design functions.

i. Review the adequacy of licensee's processing and control of operating experience information and vendor notifications

The licensee was identifying and capturing operating experience and vendor notifications.

It was noted on several occasions that the licensee was not applying vendor information correctly. Several PERS identified overthrust events on many MOV's. The licensee's engineering analysis determined that all overthrust events were acceptable based on Limitorque technical update 92-01. Limitorque technical update 92-01 allowed an increase in rating of SMB-000, SMB-00, SMB-0, and SMB-1 aqctuators to 140 percent of the current rating provided that all conditions of the technical update were met. However, the inspectors found that the licensee was applying this standard without verifying the specified conditions and also applied the analysis to actuators that were not covered (SMB-2, SMB-3, SMB-4, and SMB-5). Section 4OA4.3.l(2) documents a finding related to this issue.

j. Review MOV periodic verification test results, both static and dynamic, and verify that information from these tests are incorporated into the design and setup calculations for safety related MOVs

The licensee is a member of the Joint Owners Group (JOG) and is committed to the JOG periodic verification program for addressing GL 96-05. The licensee was in the process of implementing the recommended final JOG program. The final JOG program required participants to risk rank MOVs and classify them based on valve type, construction, materials, service conditions, manufacturer, and their susceptibility to degradation. The classification process was developed based on the five year industry test program. The process had four classification categories:

Class A: Valves are not susceptible to degradation based on test data

Class B: Valves are not susceptible to degradation based on test data and engineering analysis

Class C: Valves are susceptible to degradation as shown by test data

Class D: Valves are not covered by the JOG program. Individual plants are responsible for justifying the periodic verification approach.

The licensee has completed the classification process and has documented the results. The licensee has identified several valves that are considered to be Class D. BFN periodic verification approach for Class D valves has been to modify the valves to a known Class A or Class B valve configuration. The inspectors determined this to be an acceptable approach. However, valve modification eliminated the valve original design basis that was established to meet GL 89-10 concerns.

During GL 89-10 program inspections, the NRC staff provided four acceptable methods the licensee could use to demonstrate the design basis capability of safety-related MOVs. The four methods for demonstrating MOV capability, in descending order of acceptability were:

1. Dynamic flow testing with diagnostics of each MOV where practicable. Although the valve factor derived from the test data might be low because of minimal valve operating history or recent maintenance that exposed the stellite valve material to air, the dynamic testing provided assurance that the valve performance was predictable. The licensee considered the need to increase the valve factor during its design-basis evaluation and setup based on test data from similar valves.

2. EPRI MOV PPM. This method was developed for those valves that could not be dynamically tested. The PPM required internal measurements to provide assurance that the valve performance was predictable. The NRC staff began accepting the use of the PPM even where dynamic testing for an MOV was practicable.

3. Where valve-specific dynamic testing was not performed and the PPM was not used, the staff accepted grouping of MOVs that were dynamic tested at the plant to apply the plant-specific test information to an MOV in the group. Using plant-specific data allowed the licensee to know the valve performance and maintenance history, and helped provide confidence that the valve performance was predictable.

4. The least preferred approach (with the most margin required) was the use of valve test data from other plants or research programs because the licensee would have minimal information regarding the tested valve and its history. In such cases, the NRC inspectors would perform an available capability evaluation of the MOV to provide confidence that the MOV had significant capability margin to close GL 89-10 for that MOV.

A review of five valve modifications noted that the least preferred approach was used by the licensee for re-establishing the modified valves design basis. The licensee used similar valve test data obtained from the GL 96-05 industry test data final report. An engineering analysis compared the industry test data with the current valve set up and margin. Additionally, the engineering analysis stated that the valves were not able to be dynamically tested. The licensee used a conservative friction factor for the design basis for the new valve configuration. Although this met the minimum requirements for establishing a design basis, the extra confidence factor obtained by actual in-house dynamic testing was not being pursued by the licensee. The inspectors questioned the licensee whether the valves could be dynamically tested, a positive answer was given for most of the valves modified.

The inspectors found that during the course of developing a JOG final periodic verification approach, the JOG members noted that the effort was intended to answer the valve degradation question as it pertained to valve configuration, design, and system application. The JOG dynamic test program was not intended to provide data to the industry for the purpose of justifying valve performance. The final JOG testing approach took credit for the initial design basis developed during the GL 89-10 effort for each individual MOV. Should a valve in service have a disallowing modification, a new qualifying basis needs to be obtained. The final JOG document provided an approach for obtaining a qualifying basis for the different types of valves. The JOG approach required a certain amount of dynamic testing for reaching a qualifying basis to support the new valve configuration. Without a qualifying basis, as specified by the JOG final report, each plant was responsible for establishing a new design bases for those valves. The design basis capability approach, detailed in items 1 thru 4 above, was still applicable. Section 4OA4.3.I(1) documents a finding related to this issue.

k. Review changes made in programs affecting safety-related MOVs since the completion of the NRC review or inspection of the GL 89-10, GL 95-07 and GL 96-05 programs

A review of MOV program scope since completion of GL 89-10 and GL 95-07 was completed. The inspectors determined that the licensee was in the process of implementing the final program for meeting GL 96-05. The licensee had classified all their MOV's and had implemented many modifications to improve valve functionality and margin. Sections 4OA4.3.a and 4OA4.3.b document discrepancies identified by the NRC and the licensee related to changes made in programs affecting safety-related MOVs.

l. Findings

.1 Failure to Reestablish Motor Operated Valve Design Basis Capability after Performing Modifications to the Valves

Introduction: A finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50.55a(b)(3)(ii) was identified by the inspectors for the licensee's failure to adequately reestablish the design basis capability of multiple MOVs after internal modifications were performed to the valves.

Description: The licensee replaced internal components (valve discs) in several safety related valves, including 12-inch valves required to open for core spray injection, 20-inch valves for the RHR shutdown cooling pump suction that are required to shut for containment isolation, and 24-inch valves required to open for Low Pressure Core Injection (LPCI). Because the new internal components were not like-for-like replacements, the prior justification for the design basis capability of these MOVs was no longer valid after the valves were modified with the new components.

While reviewing design calculations for five of these valve modifications, the inspectors questioned the licensee's methodology for justifying the design basis capability of the modified valves. TVA guidance provided to the NRC inspectors specified that the following approach would be used to justify the design-basis capability for the replacement of MOVs within the Joint Owners Group (JOG) Program on MOV Periodic Verification:

1. "Perform [EPRI] PPM to determine the new design basis of COP. This is the preferred method. Implementation note: the PPM method is preferred, and provides high levels of margin as acceptable by the NRC and the industry. This method requires detailed valve design information to implement.

2. Identify similar MOV or a group of MOVs that have similar valve and system characteristics and have the established (plateau) COF [coefficient of friction]. Implementation note: This is the same methodology used in the [Generic Letter] 89-10 program and may be implemented when detailed valve design information is unavailable and sufficient test data exists. With the understanding of the JOG program, TVA will use the established COF of these groups of similar MOVs as the design basis COF or conservatively use the JOG threshold if it is greater.

3. Perform DP [Differential Pressure] testing for the MOV. Implementation note: May be performed provided sufficient testing is developed to simulate the 5-years operating time and the number of DP strokes to reach the COF plateau, this is not the preferred method."

The inspectors found that the prioritization used by the licensee for various approaches to justify the design-basis capability of MOVs did not agree with the preferred methodology outlined by the NRC in GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." Specifically, GL 89-10 stated that dynamic flow testing with diagnostic testing of the MOV is the best and preferred method for demonstrating operability. GL 89-10 further states:

> "It is also recognized that it may be impracticable to perform in situ MOV testing at design-basis degraded voltage conditions… Alternatives to testing a particular MOV in situ at design-basis pressure or flow, where such testing cannot practicably be performed, could include a comparison with appropriate design-basis test results on other MOVs, either in situ or prototype. If such test information is not available, analytical methods and extrapolations to design-basis conditions, based on the best data available, may be used until test data at design-basis conditions become available to verify operability of the MOV."

The inspectors found that for the modified MOVs in question, TVA did not apply EPRI MOV PPM that is specified as the preferred method to reestablish the design-basis capability of the MOVs. In its second preferred approach, TVA stated that test data from the JOG Program for MOV Periodic Verification, assumed to be conservative by TVA, would be used to justify the design-basis capability of similar MOVs. The NRC staff position is, unlike the EPRI MOV PPM program, the JOG program was not intended to be used in establishing the initial design-basis capability of MOVs, or modified MOVs such as at Browns Ferry. The JOG program was developed by the nuclear power plant owners groups in response to GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves." The JOG program involved repetitive dynamic testing of a small sample of MOVs under flow conditions at each participating nuclear power plant to measure any increase in valve operating requirements (valve factor for gate valves) over a 5-year period. The JOG program used the results of its testing program to identify those MOVs and their applications that could be assumed to have stable operating requirements, such that periodic dynamic testing under flow conditions was not necessary, and to specify those MOVs and their applications that needed to be periodically tested under dynamic flow conditions, to satisfy GL 96-05. Subsequently, the NRC codified the provision in GL 96-05 to establish a program to continue to maintain the design-basis capability of safety-related MOVs in 10 CFR 50.55a(b)(3)(ii).

The inspectors found that TVA Calculation Number MDQ0999980001 (Rev. 008), "MOV Calculation Input Parameters – Post JOG Implementation," in Attachment G relies on test data from two valves tested as part of the JOG Program on MOV Periodic Verification to justify the design-basis capability of the modified Walworth flexwedge gate valves at Browns Ferry. Attachment G in the TVA calculation specified that the JOG test valves were 8-inch valves with Stellite disc and guide surfaces in treated water, which TVA used as the basis for applying the JOG valve factors to the modified Browns Ferry valves. The TVA calculation relied on valve factors ranging from 0.28 to 0.44 for the two JOG test valves as justification for an assumed valve factor of 0.6 for the Browns Ferry valves. The TVA calculation did not provide adequate justification for specifying an assumed valve factor of 0.6 as conservative. For example, the TVA calculation did not address potential differences in the application and orientation of the Browns Ferry and JOG test valves, the fluid and system conditions for the Browns Ferry and JOG test valves, the operation and maintenance histories of the Browns Ferry and JOG test valves, or the wide range of valve factors demonstrated by the JOG test valves that might indicate that the JOG test valves had not achieved their plateau value for their valve factors. As a result, TVA's conclusion that the valve factor selected for the modified Browns Ferry MOVs was conservative was not justified.

In addition, the inspectors found that the TVA calculation specified that none of the Class 600 Walworth valves could be DP tested to re-establish their design basis. The NRC inspectors questioned why the valves could not be DP tested; the licensee concluded that "after considerable reviews and discussions with Browns Ferry Nuclear Engineering and SRO's [Senior Reactor Operators], we believe special DP tests instructions are feasible for the valves listed above. The DP testability statement will be removed from the calculation pages."

<u>Analysis</u>: The NRC inspectors determined that the methodology described in the TVA program documentation for justifying the design-basis capability of MOVs and the specific justification prepared by TVA to reestablish the design basis capability of MOVs after undergoing internal modifications did not satisfy 10 CFR 50.55a(b)(3)(ii) and was a

performance deficiency. This was determined to be a programmatic deficiency because there were at least 12 examples of other modified MOVs where the licensee implemented its methodology that did not provide an adequate justification for the design basis capability of those MOVs that satisfied 10 CFR 50.55a(b)(3)(ii).

The NRC staff determined the finding to be more than minor because the program deficiency, if left uncorrected, could become a more significant safety concern. Specifically, by establishing a design basis MOV valve factor that TVA considered to be conservative using data from two tested valves obtained from the JOG MOV Performance Verification program without demonstrating their applicability to the Browns Ferry motor operated valves, the established valve factor was the minimum value that must be used to set up MOV actuator operating parameters. This could lead to the incorrect assumption that there was available margin in the valve design calculations and thus allow changes to the MOV actuator operating setpoints that could result in multiple MOVs being incapable of performing their safety functions. Additionally, the licensee could incorrectly rely on perceived margin for the basis of operability of any MOV that may become degraded.

The inspectors determined this finding was associated with the Mitigating Systems Cornerstone and characterized in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone. The finding screened as having very low safety significance (Green) because the finding was a design or qualification deficiency confirmed not to result in a loss of operability at this time.

This finding had a cross-cutting aspect in the area of human performance, decision making, because the licensee failed to use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that it was unsafe in order to disapprove the action. Specifically, the licensee made modifications to multiple safety related MOVs and then reestablished their design basis capability using methods that were inconsistent with industry and NRC guidance. [H.1(b)]

<u>Enforcement</u>: 10 CFR Part 50.55a(b)(3)(ii), "Motor-Operated Valve Testing," requires, in part, that licensees shall comply with the provisions for testing MOVs in the ASME OM Code Section ISTC 4.2, 1995 Edition with the 1996 or 1997 Addenda, and shall establish a program to ensure that MOVs continue to be capable of performing their design basis safety function. Contrary to the above, over a period of 2009 through 2010, the licensee failed to implement activities that would provide assurance that specific modified MOVs were capable of performing their design basis safety functions. Specifically, BFN performed modifications to multiple safety related MOVs and failed to implement activities to provide adequate justification to reestablish the design basis capability of those modified MOVs and to ensure that they would continue to be capable of performing their design-basis safety functions, as required by 10 CFR 50.55a(b)(3)(ii). Because this violation was of very low safety significance and it was entered into the licensee's CAP as SR 457517, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy (NCV 05000259, 260, 296/2011011-03; Failure to

Reestablish Motor Operated Valve Design Basis Capability after Performing Modifications to the Valves).

.2 Inadequate Functional Evaluations Performed to Support Operability of Overthrust Motor Operated Valves

Introduction: A finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to perform an adequate review of multiple functional evaluations written to support operability of MOVs which experienced overthrust conditions.

Description: The inspectors reviewed 4 PERs related to inadvertent overthrusting of valves during testing:

- PER 156774, valve 1-FCV-074-0067
- PER 156620, valve 1-FCV-023-0046
- PER 155987, valve 1-FCV-023-0040
- PER 157250, valve 1-FCV-071-0008

For each of the above PERs, a functional evaluation was performed that relied on Limitorque Technical Update #92-01, "Kalsi Engineering Document #1707-C, Rev. 0 (11-25-91) Thrust Rating Increase SMB-000, SMB-00, SMB-0, & SMB-1 Actuators," to justify operability of the valves after being overthrusted. Technical Update #92-01 provides justification for exceeding the applied thrust of certain SMB actuators up to 140 percent of rated value, provided certain conditions are met. One of these conditions is that "the housing cover and actuator base fasteners should be torqued to the minimum levels" as outlined in the Technical Update.

The inspectors noted that 1-FCV-074-0067 had an SMB-4 actuator and 1-FCV-023-0046 and 1-FCV-023-0040 have SMB-2 actuators. Therefore, Limitorque Technical Update #92-01 was not applicable to these valves. The inspectors further noted that Valve 1-FCV-071-0008 had an SMB-0 actuator, but Browns Ferry had not torqued the actuator bolts per the Limitorque Technical Update.

The inspectors reviewed TVA procedure NEPD-22, "Functional Evaluations," which required that "a Functional Evaluation shall be performed when Site Engineering concludes that a formal documented evaluation is necessary for potentially degraded/non-conforming conditions and when requested by Operations to address operability or functionality issues." This procedure stated that "input from outside sources such as equipment vendors may be used provided consideration is given to the suitability of the source, its technical veracity, and the nature of the information provided." NEPD-22 further requires that "the responsible Manager/Supervisor shall review the documented supporting information and resolve any discrepancies with the assigned personnel." Contrary to this requirement, there were multiple instances where Limitorque Technical Update #92-01 was incorrectly used to justify operability of safety related valves and the responsible Manager/Supervisor failed to properly review the supporting information and identify these discrepancies.

Analysis: The inspectors determined that the failure to identify discrepancies with multiple functional evaluations was contrary to TVA Procedure NEPD-22, and was a performance deficiency.

The finding was determined to be more than minor because if left uncorrected, could become a more significant safety concern. Specifically, the review of the functional evaluation that is performed by the engineering supervisor is a critical part of the functional evaluation process to second check the quality of work that will ultimately be provided to the Senior Reactor Operator (SRO). The SRO will use this evaluation to aid in determining equipment operability. Therefore, the failure to thoroughly review Functional Evaluations and identify discrepancies could lead to incorrect information being used to determine equipment operability. The inspectors determined this finding was associated with the Mitigating Systems Cornerstone and characterized in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone. The finding screened as having very low safety significance (Green) because the finding was a design or qualification deficiency confirmed not to result in a loss of operability at this time.

This finding has a cross-cutting aspect in the area of human performance, decision making, because the licensee did not make safety-significant or risk-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. This included implementing these roles and authorities as designed and obtaining interdisciplinary input and reviews on safety-significant or risk-significant decisions. Specifically, on multiple occasions, the engineering supervisor responsible for reviewing an engineering evaluation that was to be provided to the SRO for a degraded equipment condition failed to implement his/her role and authority in reviewing and approving functional evaluations. The Supervisors failed to provide the interdisciplinary review during the decision making process that should have identified that a Technical Update was not applicable to safety related valve actuators. [H.1(a)]

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, 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. NEPD-22, "Functional Evaluations", required that the responsible Manager/Supervisor shall review functional evaluations and the documented supporting information, and resolve any discrepancies with the assigned personnel. Contrary to the above, in October and November of 2008, the licensee failed to accomplish activities affecting quality in accordance with a quality procedure. Specifically, on multiple occasions the licensee failed to properly review the supporting information and identify a discrepancy in a functional evaluation for motor operated valves. Because this violation was of very low safety significance and it was entered into the licensee's CAP as SR 435415, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy (NCV 05000259/2011011-03; Inadequate Functional Evaluations Performed to Support Operability of Overthrust Motor Operated Valves).

.3 IP 73756, Inservice Testing of Pumps and Valves

a. Verify that the licensee has assigned responsibilities to persons and organizations for the IST Program

The inspectors determined that licensee procedure 0-TI-362, “Inservice Testing of Pumps and Valves,” Revision 26, assigns overall responsibility for the IST program to the station IST Engineer. The IST Engineer is responsible for establishing and maintaining the BFN ASME OM Code IST Program. This includes monitoring of all IST activities by all BFN organizations and the evaluation of test results. The IST Engineer shall identify and categorize the valves within the scope of the IST program, select valve for testing at cold shutdown and refueling outage frequencies, identify pumps within the scope of the IST program, prepare requests for relief, and prepare Cold Shutdown and Refueling Outage Justifications. Procedure O-TI-362 also identifies those Class 1, 2, and 3 pumps and valves that are within the scope of the BFN IST Program.

b. Select Sample Systems to Review. Select a minimum of three ASME Code Class 1, 2, or 3 safety-related systems to review and assess for the IST of certain components in the systems

The RHR, High Pressure Coolant Injection System (HPCIS), and the RHRSWS were selected for review. The following diagrams, P&ID 1-47E812-1 for HPCIS, 1-47E858-1 for RHRSWS and 1-47E811-1 for RHR, were used in identifying what pumps and valves should be included in the IST program in accordance with the 1995 Edition through the 1996 Addenda of the ASME OM Code. These diagrams were reviewed against the BFN Technical Instruction O-TI-362, Rev. 26.

c. Verify that the pumps and valves that perform a safety-related function(s) in the selected systems are in the IST Program

The inspectors determined that there were a significant number of valves that were in the flow diagrams of RHRS, RHRSWS, and HPCIS but were not included in the BFN IST program. In response to inspector’s requests, BFN staff provided the bases and justifications for excluding certain valves from the IST program. The inspectors reviewed the corresponding responses to these SERs and PERs that incorporated these valves into the next revision of the BFN IST program and determined that the operability of these valves had been demonstrated. The following are some examples of these valves.

Among the valves reviewed, the licensee agreed that the following RHRSW valves SHV-74-91, CKV-74-674, CKV-74-698, CKV-74-706, and CKV-74-803 fitted the scope of IST program and should have been included in the BFN IST program. BFN entered them into the corrective action program in PER 431438. The inspectors reviewed the PER response that incorporated these valves into the next revision of the BFN IST program and determined that non-IST testing of these valves routinely conducted provided assurance the operability of these valves had been adequately demonstrated. By self-assessment, BFN also identified a number of valves that should be added to BFN IST program or changed from passive to active valves. BFN entered those valves into Service Request Report (SER) including SER 430035 for RHR valves SHV-74-91, FCV-74-46, and SHV-74-160; SER 430042 for Control Rod Drive valves CKV-85-806A, B, C and D, -763B, -764B, -765B, and -755B; SER 429794 for EECW valves RFV-67-783, -799, -803 and -805; SER 435240 for RHRSW valves SHV-23-512, -532, -552 and -571; and SER 430019 for RHR valves FCV-74-1, -2, -12, -13, -24, -25, -35 and -36. The inspectors reviewed the corresponding responses to these SERs and PERs that incorporated these valves into the next revision of the BFN IST program and determined that the operability of these valves had been adequately demonstrated.

Inspectors noted and the BFN staff acknowledged that since they did not have an on-site IST base document, determining the justification for including or not including certain valves in the IST program became difficult. BFN is currently preparing an IST Basis Document in conjunction with updating their IST Program to later additions of the code.

d. <u>Verify that requests for relief and approval for alternative testing have been submitted to the NRC. When the requests are not based on an impractical condition, verify that the alternative is not implemented in lieu of the code requirements prior to NRC approval</u>

The inspectors determined that the following relief requests were processed and approved. The inspectors identified two relief requests in Appendix B and Appendix C of 0-TI-362 for IST of Pumps and Valves, Revision 27. One request for relief is for Standby Liquid Control pumps (PV-1), and the other for Control Rod Drive Scram Inlet and Outlet valves (PV-2). In a letter to NRC dated September 6, 2002, TVA submitted its third ten-year IST program and the above two requests for relief from OM Code requirements. NRC staff reviewed the above two relief requests and, in a letter to TVA dated November 14, 2002, the NRC staff concluded that the proposed alternatives described in the relief requests provided acceptable level of quality and safety, and were authorized for implementation at BFN.

e. <u>Review the justification for deferring testing to cold shutdowns or refueling outages</u>

The inspectors determined that the following refueling outage justifications and Cold Shutdown Justifications were sufficient. The inspectors reviewed one Refueling Outage Justification, RO-01, for main steam valves FCV-1-14, 26, 37, and 51 for testing these valves at refueling outage frequency. The justification included that containment atmosphere was inert and remained inert during operation or most cold shutdown periods. The inspectors reviewed seven Cold Shutdown Justifications (CSJs) for Reactor Recirculation System valves FCV-68-3 and -79 (CSJ-1); Reactor Building Closed Cooling Water System valve FCV-70-47 (CSJ-2); Residual Heat Removal System valves FCV-74-47 and -48 (CSJ-3); Main Steam Out Board Isolation valves FCV-1-15, -27 -38 and -52 (CSJ-4); Main Steam Seal Supply Regulator valve 1-147 (CSJ-5); Steam Jet Air Ejector Pressure Regulatory valves 1-151, -153, -166 and -167 (CSJ-6); and Residual Heat Removal Injection valves 74-53 and -67, and Core Spray Injection valves 75-25 and -53 (CSJ-7). The inspectors reviewed the above CSJs for justification for deferral of testing above valves to the cold shutdown frequency.

f. Valve Testing: Evaluate the following areas for testing of a sample of valves in the selected systems

(1) Evaluate the adequacy of check valve testing

Inspectors observed the tests being conducted by BFN staff on check valve 0-FCV-23-502 to verify that the valve goes closed to its seat, as required by ISTC 4.5.4 of the OM Code. Procedure 2-SI-4-5-C.1, step 7.4 (11) instructs the staff to verify the closure of the check valve by verifying that the RHRSW Pump is not rotating. The inspectors questioned the adequacy of BFN's RHRSW check valve testing methodology and determined that a performance deficiency existed with BFN's methodology. The observation below describes why the inspectors determined that this performance deficiency is not a violation and that operability of the check valve is verified by other system flow characteristics.

(2) Observation

Check valves 0-CKV-23-506, 0-CKV-23-502, 0-CKV-23-522, 0-CHV23-526, 0-CKV-23-546, 0-CKV-23-542 are discharge check valves downstream of RHRSW pumps A1, A2, B1, B2, C1,and C2, respectively, and are required to be tested in accordance with the requirements of the OM Code. Section ISTC 4.5.4 requires verification that the valve goes closed and that the obturator goes to its seat.

The instructions in BFN procedure 2-SI-4-5-C.1, require the BFN staff to verify these check valve are closed by verifying that the associated RHRSW Pump is not rotating in the reverse direction. While observing testing of these check valves, the NRC inspectors questioned whether the reverse rotation of the pump was an adequate indication to verify the check valves were shut. The engineers stated that the BFN is using the guidance in NUREG 1482, Rev. 1, Section 4.1.5.5 which indicates that verifying that a centrifugal pump does not spin in the reverse direction verifies closure of a pump discharge check valve. However, there was no documented analysis, calculation, or justification to show that this method applied to the RHRSW check valves.

BFN engineers contacted the pump manufacturer, who provided information that a flow rate of between 500 to 900 gpm is required to initiate reverse rotation in the pump. BFN did not have a calculation to show that with the check valve leaking up to 900 gpm, the RHRSW system was still operable. The inspectors determined that the failure to have an adequate justification for this method of testing the check valves was a performance deficiency. BFN entered this issue into their CAP as PER 437973.

As part of the corrective action in PER 437973, BFN evaluated the conformance with OM code requirements and the operability of the check valves with the following results.

- A flow of between 500 to 900 GPM is required to initiate reverse flow in the pumps.
- During surveillance tests of the RHRSW pump in parallel with the check valve being tested a flow through the RHRSW system heat exchanger is verified to be 4500 GPM.

- Therefore the additional 900 GPM would mean the pump must be able to provide 5400 GPM and the pump curves show that the RHRSW pumps will be able to provide the additional flow.
- A flow of 900 GPM will mean that the check valve is approximately 5% off its seat.

Based on the above information BFN stated that the present test method verifies that the check valve has gone sufficiently to its seat to assure that the flow through the RHRSW system is sufficient to meet its safety function. Therefore, the OM Code requirement has been met.

This observation shows that the BFN initial review in determining compliance with OM Code requirements was insufficient. Although BFN was not explicitly taking credit for the RHRSW flow test as part of the RHRSW check valve IST test, the inspectors determined that this performance deficiency is not more than minor because the RHRSW flow test aided in verifying that the check valves go sufficiently closed to meet the valves' and the system's safety function.

Part of the corrective action in PER 437973 recognizes that additional assurance needs to be provided that the valve obturator goes to its seat, and BFN is considering other non-intrusive positive means to determine that the valve obturator goes to its seat.

g. <u>Pump Testing: Evaluate the following areas for testing of a sample of pumps in the selected systems</u>

(1) <u>Review pump testing methods, acceptance criteria, and corrective action in the test procedures</u>

The inspectors observed an IST test for RHRSW pump 2A. The test was conducted in accordance with BFN procedure 2-SI-4.5.C.1 (3), "RHRSW Pump and Header Operability and Flow Test." The inspectors observed the valve lineup and connection of temporary instrumentation. The inspectors also verified that the ranges and calibration accuracies of test instruments, and that testing was performed at established reference values. Due to problems with a security door this test was suspended and completed at a later time.

h. <u>Findings</u>

No findings were identified.

.4 <u>Follow-up of Performance Deficiencies Identified in NRC Letter ML112280215</u>

a. <u>Observations</u>

In an August 16, 2011, letter to TVA (ML112280215), The NRC discussed the Browns Ferry Red Finding and Notice of Violation. Listed in this letter are additional performance deficiencies that were identified as part of the NRC's independent review of the matter. The inspectors reviewed these performance deficiencies as part of the 95003 part 1 inspection.

(1) Inadequate Scoping of Safety Related MOV's in the Generic Letter 89-10 Program

From the August 16, 2011 Letter:

"The NRC also reviewed TVA's decision to exclude 1-FCV-74-66 from the scope of the GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," program. TVA had excluded the outboard LPCI valves from the program because it considered them to be passive valves with no safety-related function to reposition. The NRC determined that the LPCI outboard injection valves have an active safety function to close and TVA's classification was incorrect. Therefore, these valves should have been included within the scope of the GL 89-10 program. The safety functions enabled by closing these valves include several modes of post-accident containment cooling. The BFN Updated Final Safety Analysis Report stated the containment cooling function of RHR was a required safety function to mitigate an accident, and TVA's emergency operating instructions required the LPCI outboard injection valves to be repositioned closed to accomplish this function. The NRC concluded that the cause(s) for not including these valves within the scope of the GL 89-10 program was within TVA's ability to foresee and correct, and that this contributed to the performance deficiency."

The inspectors determined that this performance deficiency is an example of a failure to establish an adequate program to provide assurance that MOVs continue to be capable of performing their design basis safety function by scoping safety related MOV's in the GL 89-10 program. Because this performance deficiency is directly related to the more broadly stated performance deficiency in the August 16, 2011, letter, there was no additional finding associated with this performance deficiency. TVA wrote PER 424419 to track adding valves FCV-74-52 and FCV-74-66 in BFN's GL 89-10 MOV Program.

While reviewing this performance deficiency, the NRC inspectors identified an additional performance deficiency in that TVA did not provide complete and accurate information to the NRC in a signed letter dated January 6, 1997, and also in a signed letter dated May 5, 2004. This issue is further discussed in section 4OA4.5.b(1).

(2) Inadequate Acceptance Criteria Contained in Procedures for Performing Partial MOVATS Testing

From the August 16, 2011 Letter:

"The NRC assessed TVA's review of the partial Motor Operated Valve Analysis and Test System (MOVATS) testing (which included a time trace of electrical current taken at the motor control center) performed on valve 1-FCV-74-66 on October 31, 2008. This partial MOVATS testing provided evidence that the valve's disc was detached from the stem. The NRC determined that a more comprehensive review of the test data by TVA would likely have resulted in a more timely identification of the stem to disc separation. In addition, Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that safety-related procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been successfully accomplished. TVA's Procedure ECI-0-000-MOV009, "Testing of Motor Operated Valves using MOVATS Universal Diagnostic System (UDS) and Viper 20," Revision 20, a safety-related procedure, did not contain appropriate quantitative or qualitative acceptance criteria for determining that partial MOVATS testing was successfully accomplished. TVA's failure to include appropriate quantitative or qualitative acceptance criteria for partial MOVATS testing in Procedure ECI-0-000-MOV009 was within its purview and contributed to the performance deficiency."

The inspectors determined that this performance deficiency is an example of a failure to establish an adequate program to provide assurance that MOVs continue to be capable of performing their design basis safety function by including appropriate acceptance quantitative or qualitative criteria for determining that partial MOVATS testing was successfully accomplished. Because this performance deficiency is directly related to the more broadly stated performance deficiency in the August 16, 2011, letter, there was no additional finding associated with this performance deficiency. TVA wrote PER 431350 to address the performance deficiency for including acceptance criteria in BFN procedures.

(3) <u>Failure to Identify a Condition Adverse to Quality while Venting 1-FCV-74-66</u>

While reviewing prior correspondence between the NRC and TVA contained in NRC file memo dated August 4, 2011, "Submittal of Reference Documents Related to EA-11-018 for Browns Ferry Nuclear Plant Unit 1, Docket 50-259" (ML 11216A118), the inspectors noted that TVA missed an opportunity to identify a condition adverse to quality with FCV-74-66.

TS Surveillance Requirement (SR) 3.5.1.1 (RHR I and RHR II) states, "verify, for each ECCS [Emergency Core Cooling System] injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve." This SR is required to be accomplished monthly and is normally performed by opening a high point vent line to allow any entrapped air to escape the system. When a continuous steady stream of water is observed leaving the vent line, the vent valve is shut. On November 11, 2008, operators performing the monthly SR were unable to obtain a continuous steady stream of water while venting the bonnet of valve 1-FCV-74-66. This was the third performance of the venting evolution, and the previous two performances were completed successfully.

Operators and engineers thought an obstruction in the vent line was the cause for the unsuccessful completion of SR 3.5.1.1. Work Order (WO) 08-723810-000 was written to clear the obstruction that was thought to be in the vent line. Craft personnel performed the WO but were unable to clear any obstruction and successfully vent the RHR loop II line. PER 156971 was then written to document the failure to vent the bonnet of FCV-74-66 and clear any obstruction from the vent line.

WO 08-723813-000 was written to perform troubleshooting and venting of the line, or to replace the valve and line to re-establish the vent path if troubleshooting was unsuccessful. Functional Evaluations 42924 and 43012 were written to provide technical justification for the non-conforming condition and to require a revision to the SR 3.5.1.1 procedure to perform ultrasonic testing of the RHR line to verify that it was full of water. Ultrasonic testing was considered an acceptable method to meet the requirements of SR 3.5.1.1. WO 08-723813-000 was completed on November 10, 2010, when a prefabricated assembly consisting of a new vent line and vent valve was used to replace the installed vent line and valve. WO 08-723813-000 did not include a step to examine the removed vent line and valve to determine if an obstruction actually existed. Because TVA did not verify an obstruction existed in the vent line, it is possible that the failure to vent the line was due to the stem disc separation of FCV-74-66.

The inspectors determined that the failure to thoroughly evaluate the condition adverse to quality as to why operators were unable to vent the FCV-74-66 valve on November, 11, 2008, was a failure to identify and correct a condition adverse to quality, and was therefore a performance deficiency. Specifically, had BFN evaluated all possible causes of the failure to vent FCV-74-66, personnel may have identified that the failure to vent was possibly due to a stem disc separation of valve FCV-74-66 and not an obstruction in the vent line.

The inspectors determined that this performance deficiency is an example of a failure to establish an adequate CAP to provide assurance that MOVs continue to be capable of performing their design basis safety function. Because this performance deficiency is directly related to the more broadly stated performance deficiency in the August 16, 2011, letter, there was no additional finding associated with this performance deficiency. TVA wrote SR 435769 to address this performance deficiency as part of their revised Root Cause Analysis for the Red finding. This SR will specifically look at troubleshooting techniques, management oversight, and rigor of technical review as applied to the failure to identify and correct a condition adverse to quality related with the venting of valve FCV-74-66.

b. Findings

(1) Inaccurate Information Provided Regarding Scoping of Motor Operated Valves in the Generic Letter 89-10 Program

Introduction: While reviewing information pertaining to the removal of the Unit 1, 2, and 3 FCV-74-52 and FCV-74-66 valves from BFN's GL 89-10 program, the inspectors identified that TVA had provided incomplete and inaccurate information to the NRC in a letter from T. E. Abney, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 89-10, Safety Related Motor-Operated Valve (MOV) testing and Surveillance, NRC Inspector Follow-up Item (IFI) 50-260, 296/95-19-01, Response to Request for Reevaluation Regarding Scope of MOVs," dated January 6, 1997.

Additionally, TVA again provided incomplete and inaccurate information to the NRC in a letter from T. E. Abney, "Browns Ferry Nuclear Plant (BFN) Unit 1 – Generic Letter 89-10 and Supplements 1 to 7, Safety-Related Motor-Operated Valve (MOV) Testing and

Surveillance," dated May 5, 2004. This was an apparent violation of 10 CFR 50.9, "Completeness and Accuracy of Information."

Description: In late 1994, TVA reevaluated the safety functions of the MOVs in the GL 89-10 program and subsequently removed 20 MOVs from the scope of the program. TVA removed these valves after it concluded that the MOVs did not have an active safety function.

To determine why BFN had not scoped MOV 1-FCV-74-66, RHR Loop II Outboard LPCI Throttle Valve, into the station's GL 89-10 program, the inspectors reviewed letters between TVA and the NRC. In a letter from Paul E. Fredrickson to Oliver Kingsley, "Inspector Follow-up Item 50-260/95-19-01 and 50-296/95-19-01 Reduced Scope of Valves in Generic Letter 89-10 Program," dated October 7, 1996, the NRC documented an assessment of the reduction in scope of MOVs included in BFN's GL 89-10 Program. The NRC asked BFN to reexamine the safety functions of their MOVs consistent with the information provided in the NRCs assessment and to provide a response to the NRC along with any appropriate corrections to the 89-10 program. In the letter, the NRC stated that "FCV-74-52 and 74-66 appear to have a redundant safety function with FCV-74-53 and FCV-74-67, respectively, to close to allow operation of the suppression pool cooling mode of the RHR system."

TVA responded to the NRC in a letter from T. E. Abney, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 89-10, Safety Related Motor-Operated Valve (MOV) testing and Surveillance, NRC Inspector Follow-up Item (IFI) 50-260, 296/95-19-01, Response to Request for Reevaluation Regarding Scope of MOVs," dated January 6, 1997. In this response, TVA stated that "closure of Valves FCV-74-52 and FCV-74-66 is not required by plant procedures to operate the RHR system in the suppression pool cooling mode. Therefore, these valves have no "redundant" safety function and will not be included in the GL 89-10 program."

While reviewing station procedures, the inspectors determined that the MOV 1, 2, and 3-FCV-074-52 and FCV-74-66 are required to be shut by plant procedures, including the suppression pool cooling mode of RHR. The following procedures are examples that all require shutting or throttling FCV-74-52 and FCV-74-66:

- EOI Appendix-17A, RHR System Operation Suppression Pool Cooling
- EOI Appendix-17B, RHR System Operation Drywell Sprays
- EOI Appendix 17C, RHR System Operation Suppression Chamber Sprays
- EOI Appendix 17D, RHR System Operation Shutdown Cooling
- EOI Appendix-6B, Injection Subsystem Lineup RHR System I LPCI Mode

The inspectors determined that the information provided by TVA to the NRC in the January 6, 1997, letter stating that closure of valves FCV-74-52 and FCV-74-66 was not required by plant procedures to operate the RHR system in the suppression pool cooling mode was incomplete and inaccurate. Additionally, the inspectors identified a letter from T. E. Abney, "Browns Ferry Nuclear Plant (BFN) Unit 1 – Generic Letter 89-10 and Supplements 1 to 7, Safety-Related Motor-Operated Valve (MOV) Testing and

Surveillance," dated May 5, 2004, that also contained incomplete and inaccurate information. The May 5, 2004, letter, which was written to provide an updated response to NRC GL 89-10 for Unit 1, contained inaccurate information regarding the safety position of the 1-FCV-74-52 and 1-FCV-74-66 valves. Additionally, this letter referenced the January 6, 1997, TVA letter which contains inaccurate information regarding Units 2 and 3. When the information contained in the letters is viewed as a whole, TVA conveyed to NRC that Unit 1 had the same MOV issues in Unit 1 as Units 2 and 3.

On September 21, 2011, the inspectors discussed this issue with the licensee. On September 22, 2011, BFN notified the NRC per 10 CFR 50.9(b) that it had provided inaccurate and incomplete information in the January 6, 1997 letter. This issue was added to BFN's CAP as PER 436583 for long term evaluation and corrective action.

Analysis: The inspectors determined that the failure to provide complete and accurate information to the NRC was contrary to the requirements of 10 CFR 50.9, and was an apparent violation. Because violations of 10 CFR 50.9 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the NRC Enforcement Policy.

Enforcement: Title 10 CFR 50.9 requires, in part, that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

Contrary to the above, on January 6, 1997, TVA provided information to the Commission that was not complete and accurate in all material respects. Specifically, in a letter dated October 7, 1996, the NRC asked TVA to further consider and respond to questions about whether valves FCV-74-52 and FCV-74-66 had a redundant safety function to close to allow operation of the suppression pool cooling mode of the RHR System. In a letter dated January 6, 1997, TVA responded that "Closure of valves FCV-74-52 and FCV-74-66 is not required by plant procedures to operate the RHR system in the suppression pool cooling mode. Therefore, these valves have no 'redundant' safety function and will not be included in the GL-89-10 program." This information was inaccurate because the FCV-74-52 and FCV-74-66 valves do have a safety function to shut to operate the RHR system in the suppression pool cooling mode as described in EOI Appendix-17A, "RHR System Operation Suppression Pool Cooling," and should therefore have been included in Browns Ferry's GL 89-10 MOV monitoring program.

Additionally, The NRC identified that incomplete and inaccurate information was also provided in a letter dated May 5, 2004. This letter stated that "TVA's review and documentation of the design basis for the operation of each Unit 1 MOV within the scope of the GL 89-10 program, the methods for determining and adjusting its switch settings, testing, surveillance and maintenance are the same as with the Units 2 and 3 program."

This information was material to the NRC because it was used, in part, as the basis for determining that valves FCV-74-52 and FCV-74-66 did not meet the conditions necessary that would require them to be in Browns Ferry's GL 89-10 MOV monitoring program. The issue was preliminarily determined to be an apparent violation of 10 CFR 50.9 (AV 05000259, 260, 296/2011011-02; Inaccurate Information Provided Regarding Scoping of Motor Operated Valves in the Generic Letter 89-10 Program). This issue was entered into BFN's CAP as SR 435463, "95003 – PER 430439 documented that Units 1, 2, and 3 RHR Outboard Injection Valves, FCV-74-52 and FCV-74-66, Should have been Included in the Scope of the GL 89-10 Program. The Purpose of this SR is to Assess the Technical Basis and Adequacy of the NRC Correspondence for the GL 89-10 Scope removal of These Valves in the mid 1990's."

(2) Unresolved Item: Verification of Valve Obturator as Required by ASME OM Code

Introduction: An Unresolved Item (URI) was identified by the NRC Inspectors related to implementation of the ASME OM Code for verification of valve obturator movement.

Description: NRC Inspection Report 05000259/2011008 (ML111290500) documents additional reviews conducted regarding the adequacy of the IST program at BFN. Specifically, the NRC documented that "TVA's failure to implement in IST program in accordance with the American Society of Mechanical Engineer (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addendum, Section ISTC 4.1, precluded the timely identification that the RHR Loop II subsystem was unable to fulfill its safety function due to a failure of the LPCI Outboard Injection Valve 1-FCV-74-66. The NRC concluded that TVA's IST program inadequacy was well within its purview, and represents a performance deficiency. Details of the NRC's final determination regarding the performance deficiency are discussed in Enclosure 2."

In a letter dated June 8, 2011, TVA appealed the final significance determination of the Red finding. This letter indicated that other licensees understand and implement ASME Operation and Maintenance Code Section ISTC 4.1 in a similar manner to TVA. The NRC recognized the potential generic implications associated with this issue.

In the August 16, 2011, Letter to TVA, the NRC states:

"With respect to the IST performance deficiency described in our May 9 inspection report, the NRC determined that the requirements of the ASME OM Code concerning the verification of valve obturator position warrant additional clarification due to the diversity of views among NRC staff and industry experts. As a result, the NRC staff will continue to pursue generic resolution of the OM code testing issues separate from the resolution of this finding."

Until the ASME OM Code testing issues are resolved and there is clarification and guidance on the requirements of ASME OM Code Section ISTC 4.1, this issue is considered an URI. Once resolution has been determined, TVA's IST program will be re-evaluated to determine whether it met ASME OM Code Section ISTC 4.1 requirements and whether a performance deficiency exists or not. (URI 05000259/2011011; Verification of Valve Obturator as Required by ASME OM Code)

4OA6 Exit Meeting

On October 3, 2011, inspection team presented the inspection results to Mr. Preston Swafford, Tim Cleary, Keith Paulson and other members of your staff. The inspectors confirmed with the licensee that no proprietary information was reviewed by the inspectors during this inspection period and no proprietary information was therefore retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

B. Baker, BFN Operations Supervisor Support
C. Boschett, Quality Assurance Manager
S. Brown, BFN Project Director
B. Byrne, BFN Security Manager
T. Chan, Corporate MOV Program
T. Cleary, Vice President
P. Donnahue, BFN Asistant Engineering Director
M. Durr, BFN Engineering Director
C. J. Gannon, BFN Plant Manager
M. Gowen, Corporate Programs
G. Hall, Station Human Performance Manager
P. Herrmann, BFN Licensing Program Manager
L. Hughes, Operations Manager
R. Krich, Vice President of Nuclear Licensing
M. Oliver, Licensing Engineer
B. Pierce, BFN Performance Improvement Manager
K. Polson, Site Vice President
E. Quidley, 95003 Root Cause Lead
J. Williams, General Manager TVA Engineering

LIST OF ITEMS OPENED AND CLOSED

Opened

05000259, 260, 296/2011011-04	AV	Inaccurate Information Provided Regarding Scoping of Motor Operated Valves in the Generic Letter 89-10 Program (Section 4OA4.5.b(1))
05000259, 260, 296/2011011-05	URI	ASME Code Compliance PD (Section 4OA4.5.b(2))

Opened and Closed

05000259, 260, 296/2011011-01	NCV	Failure to Implement Requirements of the Inservice Testing Program (Section 4OA4.2.d(1))
05000259, 260, 296/2011011-02	NCV	Failure to Reestablish Motor Operated Valve Design Basis Capability after Performing Modifications to the Valves (Section 4OA4.3.l(1))

05000259/2011011-03	NCV	Inadequate Functional Evaluations Performed to Support Operability of Overthrust Motor Operated Valves (Section 4OA4.3.l(2))

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Problem Evaluation Reports (PER) and Service Requests (SR) Reviewed

PER 122218; Stem failure of 2-FCV-023-0052
PER 141380; U3 RHRSW Outlet Valve Damage Evaluation
PER 161379; Unplanned Entry into an LCO
PER 220807; Motor Leads Reversed to BFN-3-FCV-074-0053
PER 225690; Metal Shavings Discovered in Grease
PER 271338; 1-FCV-074-0066 RHR Outboard (Loop 2) Valve Failure
PER 275701; Anomalies noted during MOVATS Testing on 1-FCV-023-0052
PER 303097; No Anti-rotation Keyway Installed on 1-FCV-074-0066
PER 303100; UT of 2-FCV-074-0052 Valve Stem
PER 303497; UT of 2-FCV-074-0052 Valve Stem
PER 390118; C1 RHRSW Pump Failed Surveillance on Low Flow
PER 155882; 3-FCV-074-0066 Possibly Throttled Below 30%
PER 156926; Spring Pack Cavities Full of Grease
PER 424419; This Per is to track adding valves 74-52 and 74-66 into the GL 89-10 Program
PER 329281; Attempt to engage declutch lever in MVOP-074-0046
PER 156926; Spring Pack Cavities full of Grease
PER 142985; 3-MVOP-023-0046 Motor Pinion Key Failure
SR 109308; RHRSW/EECW IST Pump Test Non-Conformance
PER 157777; MOV PERs related to Unit 1 Restart Workmanship
PER 156774; 1-FCV-074-0067 found Overthrust During As-Found MOVATS Testing
SR 433858; 95003 NRC identified issue on engineering evaluation of PER 156774
PER 157250; 1-FCV-071-0008 Over-Thrust
PER 285603; Operability and Functionality Determination Weaknesses
PER 434900; NRC Identified – 95003 – Identified an issue with the Coefficient of Friction (COF)
PER 436583; 95003 – GL 89-10 Valve Program Scoping Submittal
SR 435769; 95003 Root Cause Team area of consideration for venting issues in RCA 369800
SR 435415; NRC Identified 95003 – Overthrust evaluations on MOV's
SR 435463; 95003 – GL 89-10 valve program scoping submittal
PER 430431; There is no document that interfaces between Appendix R, IST, MOV an dPRA Risk Ranking of vlv 95003
PER 424419; This PER is to track adding valves 74-52 and 74-66 into the GL 89-10 Program
PER 431350; As part of the review for 95003 Root Cause PER 369800, the following issue was identified
PER 430439; 95003 Inspection Readiness Review – IST – Item 2

PER 156971; Inability to vent RHR loop II on Unit 1
SR 435415; NRC Identified 95003 – Overthrust evaluations on MOVs
SR 457517; 95003 – NRC Identified a Non-Conformance Issue with the Implementation of the JOG Program
PER 431434; HPCI Relief Valves should be added to the IST Program
PER 430037; Relief Valves are not scoped into the Augmented IST Program
PER 430439; RHR Passive Valves should be changed to active valves.
PER 431395; EECW Relief Valves are not currently in the IST Program
PER 431 438; RHR Manual Valves not scoped in the IST Program
PER 437973; RHRSW check valve testing
PER 431 415; RHR, Core Spray, RCIC, HPCI, check valves not scoped in the IST Program
SR 430028; CAD system valves should be added to the Augmented IST Program
SR 430035; RHR Manual Valves are not scoped into the IST Program
SR 430042; Check valves are not scoped into the IST Program
SR 435250; NRC identified passive valves that should be added to the IST Program

Work Orders (WO)

WO 111673268; 2-SR-3.5.1.6(CS II) – Core Spray Flow Rate Loop II
WO 112212914; BFN-3-FCV-074-0066 RHR Loop II LPCI Outboard Injection Valve
WO 08-718412; BFN-3-FCV-074-0066 RHR Loop II LPCI Outboard Injection Valve Disassemble
WO 112119451; 2-SR-3.5.1.6(CS I) - Core Spray Flow Rate Loop I
WO 111674058; 2-SR-3.5.1.6(CS II) – Core Spray Flow Rate Loop II
WO 110757439; 2-MVOP-071-0038, RCIC System CNDS Test Valve overthrusted during as-found testing
WO 08-723810-000; During Performance of 1-SR-3.5.1.1 (RHR II), at step 7.2[6], unable to vent at the 1-FCV-74-66 bonnet vent
WO 08-723813-000; While Attempting to vent per 1-SR-3.5.1.1(RHR II), SHV-74-626B and VTV-74-627B Yielded only ~2 cups
112243526; PM #P2352A – HPCI Main and Booster Pump Set Developed Head & Flow Rate test at rated reactor pressure
111433752; Core Spray Loop comprehensive pump test

Calculations:

MDQ1-074-2002-0074 Calc 1-FCV-074-0057 and 1-FCV-074-0071
MDQ1-074-2002-0051 Calc 1-FCV-074-0048
MDQ1-074-2002-0075 Calc 1-FCV-074-0053 and 1-FCV-074-0067]

MOV Design Documents

MD-Q0999-910034 NRC Generic Letter 89-10 MOV Evaluation
DS-M18.2.21 MOV Thrust and Torque Calculations
DS-M18.2.22 MOV Design Basis and JOG Review Methodologies
NETP-115 MOV Program
O-TI-362 Inservice Testing of Pumps and Valves
Various Vendor Valve Weak Link Calculations

O-TI-444 Augmented Inservice Testing Program
O-TI-443 Condition Monitoring of Check Valves

Procedures:

0-TI-230V; Vibration Program; Revision 8
0-TI-362; Inservice Testing of Pumps and Valves; Revision 27
2-SI-4.5-C.1(3); RHRSW Pump and Header Operability and Flow Test; Revision 114
2-SR-3.5.5.5; RCIC System Rated Flow at Normal Operating Pressure; Revision 53
2-SR-3.5.1.6(CS II); Core Spray Flow Rate Loop II; Revision 9
2-SR-3.5.1.6(CS I); Core Spray Flow Rate Loop I; Revision 28
NEPD-22; Functional Evaluations; Revision 0009
OPDP-8; Limiting Conditions for Operation Tracking; Revision 0005
NPG-SPP-06.3; Pre/Post Maintenance Testing, Rev. 0
0-TI-362; Inservice Testing of Pumps and Valves, Rev. 26
SI-3.1.12; HPCI Baseline Data Evaluation, Rev. 9
O-TI-577; Inservice Testing of Safety and Relief valves, Rev. 00
O-TI-444; Augmented Inservice Testing Program
O-TI-443; Condition Monitoring of Check Valves, Rev. 6
O-TI-383; Evaluation of Test Results for ASME OM Code IST Program, Rev. 01

Miscellaneous Documents:

LER 50-249/2010-004; Residual Heat Removal System Pump Motor Failure; Revision 1
Browns Ferry NP Trending Report of U1C7; March 14, 2009
Limitorque Technical Update #92-01; Kalsi Engineering Document #1707-C, Rev. 0 (11-25-90) Thrust Rating Increase SMB-000, SMB-00, SMB-0, SMB-1 Actuators
Letter from Frederick J. Hebdon to Jon R. Johnson; “Task Interface Agreement 95-12 – Browns Ferry nuclear Plant Units 2 and 3 Generic Letter 89-10 Scope Change (TAC Nos. M93580 and M93581)”; August 29, 1996
Letter from Paul Frederickson to Oliver Kingsley, Jr; “Inspector Follow-up Item 50-260/95-19-01 and 50-296/95-19-01 Reduced Scope of Valves in Generic Letter 89-10 Program”; October 7, 1996
Letter from T. E. Abney to U.S. Nuclear Regulatory Commission; “Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 89-10 Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance, NRC Inspector Follow-up Item (IFI) 50-260, 296/95-19-01, Response to Request for Reevaluation regarding Reduced Scope of MOVs”; January 6, 1997
Letter from T. E. Abney to U.S. Nuclear Regulatory Commission; “Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 89-10 Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance, Program Closure (TAC Nos. M75636 and M75637); December 15, 1997
Letter from Mark S. Lesser to O. J. Zeringue; “NRC Integrated Inspection Report 50-259/97-11, 50-260/97-11, 50-296/97-11, Notice of Violation an Notice of Deviation”; January 2, 1998
Letter from T. E. Abney to U.S. Nuclear Regulatory Commission; “Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Generic Letter (GL) 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves, request for Additional Information (TAC Nos. M97021 and M97022)”; March 30, 1999

Drawings:

Drwg. 1-47E811-1-ISI; Unit 1 ASME Section XI RHR System Code Class Boundaries
Drwg. 2-47E811-1-ISI; Unit 2 ASME Section XI RHR System Code Class Boundaries
Drwg. 3-47E811-1-ISI; Unit 3 ASME Section XI RHR System Code Class Boundaries
Drwg. 1-47E812-1-ISI; Unit 1 ASME Section XI HPCI System Code Class Boundaries
Drwg. 2-47E812-1-ISI; Unit 2 ASME Section XI HPCI System Code Class Boundaries
Drwg. 3-47E812-1-ISI; Unit 3 ASME Section XI HPCI System Code Class Boundaries
Drwg. 1-47E858-1-ISI; Unit 1 ASME Section XI RHRSW System Code Class Boundaries
Drwg. 2-47E858-1-ISI; Unit 2 ASME Section XI RHRSW System Code Class Boundaries
Drwg. 3-47E858-1-ISI; Unit 3 ASME Section XI RHRSW System Code Class Boundaries

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ASME	American Society of Mechanical Engineers
BFN	Browns Ferry Nuclear Plant
CAP	Corrective Action Program
CFR	Code of Federal Regulations
COF	Coefficient if Friction
CSJ	Cold Shutdown Justification
DP	Differential Pressure
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
GL	Generic Letter
ICR	Institute of Conflict Resolution
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IST	Inservice Testing
JOG	Joint Owners Group
LPCI	Low Pressure Coolant Injection
MOV	Motor Operated Valve
MOVATS	Motor Operated Valve Analysis and Test System
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
OE	Operating Experience
OM Code	Code for Operation and Maintenance
PARS	Publicly Available Records System
PEC	Pre-Decisional Enforcement Conference
PER	Problem Evaluation Report
PPM	Performance Prediction Methodology
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
ROP	Reactor Oversight Process
SDP	Significance Determination Process

SOE	System Operational Enhancement
SR	Service Request or Surveillance Requirement
TBD	To Be Determined
TS	Technical Specification
TVA	Tennessee Valley Authority
UDS	Universal Diagnostic System
URI	Unresolved Item
WO	Work Order