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June 8, 2011

10 CFR 2.201

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Browns Ferry Nuclear Plant, Unit 1
Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Appeal of Final Significance Determination of a Red Finding and Reply to a Notice of Violation; EA-11-018**

References:

1) Letter from NRC to TVA, "Final Significance Determination of a Red Finding, Notice of Violation, and Assessment Follow-up Letter (NRC Inspection Report No. 05000259/2011008) Browns Ferry Nuclear Plant," dated May 9, 2011

2) Letter from NRC to TVA, "NRC Report 05000259/2010005, 05000260/2010005, and 05000296/2010005; Preliminary Greater Than Green Finding Browns Ferry Nuclear Plant," dated March 2, 2011

3) Letter from TVA to NRC, "Request for Regulatory Conference or Public Management Meeting," dated June 2, 2011

4) Letter from NRC to TVA, "Response to Request for Regulatory Conference or Public Management Meeting," dated June 7, 2011

Reference 1 provided the Nuclear Regulatory Commission (NRC) Final Significance Determination (FSD) and a Notice of Violation (NOV) related to the degraded condition of the Browns Ferry Nuclear Plant (BFN), Unit 1, low pressure coolant injection (LPCI)/residual heat removal (RHR) outboard injection valve 1-FCV-74-66. The valve failed to open on October 23, 2010, when operators attempted to place RHR Shutdown Cooling loop II in service to support Unit 1 refueling outage activities. The valve disc was subsequently found to be separated from the disc skirt/stem assembly and lodged in the seat. The NRC's finding was that the licensee's failure to establish adequate design control and perform adequate maintenance on valve 1-FCV-74-66, which resulted in the valve being left in a significantly degraded condition and RHR loop II unable to fulfill its safety function, was the originally identified performance deficiency. The NRC had provided a preliminary greater than Green significance determination in Reference 2. The NRC concluded in its letter of May 9, 2011 that the FSD for the valve failure was Red due primarily to fire risk. In its May 9, 2011 letter, the NRC also acknowledged that inadequate design control may not have been a primary contributor to the valve failure and that it was unlikely that the valve failure was caused by unthreading of the valve internals due to undersized welds.

The Tennessee Valley Authority (TVA) acknowledges the safety significance of this event and continues to implement extensive corrective actions to reduce fire risk as quickly as possible at BFN. As an example, BFN has just completed implementing changes to certain of the Safe Shutdown Instructions that can be used in the event of an Appendix R type fire to provide guidance to operators on the use of alternate means to provide water to the reactor core should the primary means not be available. TVA is continuing to implement the changes discussed at the April 4, 2011 Regulatory Conference and illustrated by the graph provided in Enclosure 1.

While TVA does not agree with the performance deficiency identified in the May 9, 2011 NRC letter regarding the non-compliance of BFN's Inservice Testing (IST) Program with the American Society of Mechanical Engineers (ASME) Operation and Maintenance Code (OM Code), 1995 Edition, 1996 Addendum, Section ISTC 4.1, TVA has initiated a number of actions to enhance the IST program related to the issues raised in Reference 1. Most significantly, TVA has assembled a team of industry-recognized IST experts including the principal author of NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Revision 1. This team is conducting a review of BFN's IST Program and its implementation for compliance and performance issues. The IST programs for TVA's other nuclear plants will likely be included in this review process. Other corrective actions may be identified once the root cause analysis is completed and approved.

TVA has also been in contact with other licensees of similar plants and pressurized water reactor plants regarding their implementation of OM Code Section ISTC 4.1 and found that their understanding and implementation of this code requirement is the same as TVA's. Accordingly, TVA has engaged the Nuclear Energy Institute (NEI) on the need to assemble an industry perspective with regard to this issue and consider the generic implications of implementing the requirements of the OM Code as described in the May 9, 2011 NRC letter and whether any appropriate industry actions are warranted.

As demonstrated during the Regulatory Conference on April 4, 2011, TVA thoroughly analyzed the safety significance, root cause, and extent of cause and condition of the failure of valve 1-FCV-74-66. When disassembled following its failure, valve 1-FCV-74-66 was found to have several non-conformances relative to applicable design documents, including lack of an anti-rotation key, undersized tack welds between the disc and disc skirt/stem assembly, and lack of a thrust washer. These non-conformances were the basis for the original performance deficiency identified by the NRC in Reference 2, because they were assumed to be the cause of the disc separation from the disc skirt/stem assembly. However, the actual cause of the failure was found to be a manufacturing deficiency, i.e., undersized threads in the disc skirt, which resulted in axial separation of the disc from the disc skirt/stem assembly. The originally identified deficiencies were found to not be contributors to the valve failure as acknowledged in the NRC's May 9, 2011 letter. Because TVA could not have reasonably identified the manufacturing deficiency, no licensee performance deficiency existed. Corrective actions have been taken to assure operability of valve 1-FCV-74-66 and identical valves at BFN.

Subsequent to the Regulatory Conference, the NRC identified an inadequate IST program at BFN as the performance deficiency in Reference 1, where the NRC stated that the required IST program provided TVA with an opportunity to preclude and/or identify the LPCI valve failure sooner, and concluded that certain aspects of BFN's IST Program were inadequate. TVA considers that BFN's IST Program and its implementation are in compliance with Section ISTC 4.1 of the OM Code as it is understood by TVA and many others in the industry, and that no licensee performance deficiency related to IST was related to the valve failure. This issue was not addressed by TVA at the Regulatory Conference since the NRC did not, prior to the conference, identify the IST program as an issue. The NRC's position in the FSD regarding the IST program is not based on a complete understanding of the issues since TVA did not have an opportunity to address the matter during the Regulatory Conference. As a result, TVA's June 2, 2011 letter requested another Regulatory Conference or public management meeting to present information that was not provided at the Regulatory Conference nor in answer to NRC questions before and after the Regulatory Conference (Reference 3). As stated in our June 2, 2011 letter, TVA has assembled new information provided in Enclosure 2 regarding BFN and industry implemented IST programs that are pertinent to the performance deficiency identified in the NRC's FSD. As a result, TVA has concluded as detailed in Enclosure 2, that there were no methods other than the ones implemented by TVA to comply with the OM Code to detect the failure of valve 1-FCV-74-66. The NRC responded to our letter on June 7, 2011 (Reference 4), and denied TVA's request for a forum to discuss the performance deficiency identified as BFN's IST Program. The NRC's June 7, 2011 letter also denied TVA's request to delay the date by when an appeal of the FSD must be submitted.

Accordingly, TVA has carefully reviewed the FSD and NOV and provides this response. TVA respectfully appeals the FSD in accordance with the process established in NRC Inspection Manual Chapter 0609, Attachment 0609.2, "Process for Appealing NRC Characterization of Inspection Findings." Additionally, in accordance with NRC Manual Chapter 0305, Section 12.06, TVA requests that NRC approve a deviation from the Action Matrix with respect to this finding. The detailed bases for the FSD appeal and Action Matrix deviation are provided in the Enclosure 2. This deviation would be related to the scope of the NRC inspection that will be

conducted in accordance with NRC Inspection Procedure 95003, and would be the subject of a meeting between the NRC and TVA as noted in the NRC's June 7, 2011 letter. Also, TVA's reply to the NOV is provided in Enclosure 3.

This appeal has sufficient merit for review since it falls into the category of new information that was not available at the time of the Regulatory Conference. As explained above, this information was not available at the time of the Regulatory Conference because TVA did not know at that time that an inadequate IST program would be identified as the performance deficiency. TVA informed the NRC in its June 2, 2011 letter, within the 30-day appeal period, that new information had been assembled. Furthermore, this performance deficiency raises significant generic issues because BFN's IST Program for Section 4.1 of the OM Code is consistent with the IST programs for this provision of the code at other nuclear plants based on a preliminary survey of 17 other nuclear plants. As a result, NEI is now looking into the generic implications of implementing the OM Code as described in the NRC's May 9, 2011 FSD letter.

There are no new regulatory commitments in this FSD appeal. Should you have any questions, please contact Rod M. Krich at (423) 751-3628.

Respectfully,

Preston D. Swafford

Preston D. Swafford

Enclosures:

1) Fire SDP Risk Reduction Curve
2) Final Significance Determination - Appeal, Cornerstone: Mitigating Systems
3) Reply to a Notice of Violation; EA-11-018

cc (Enclosures):

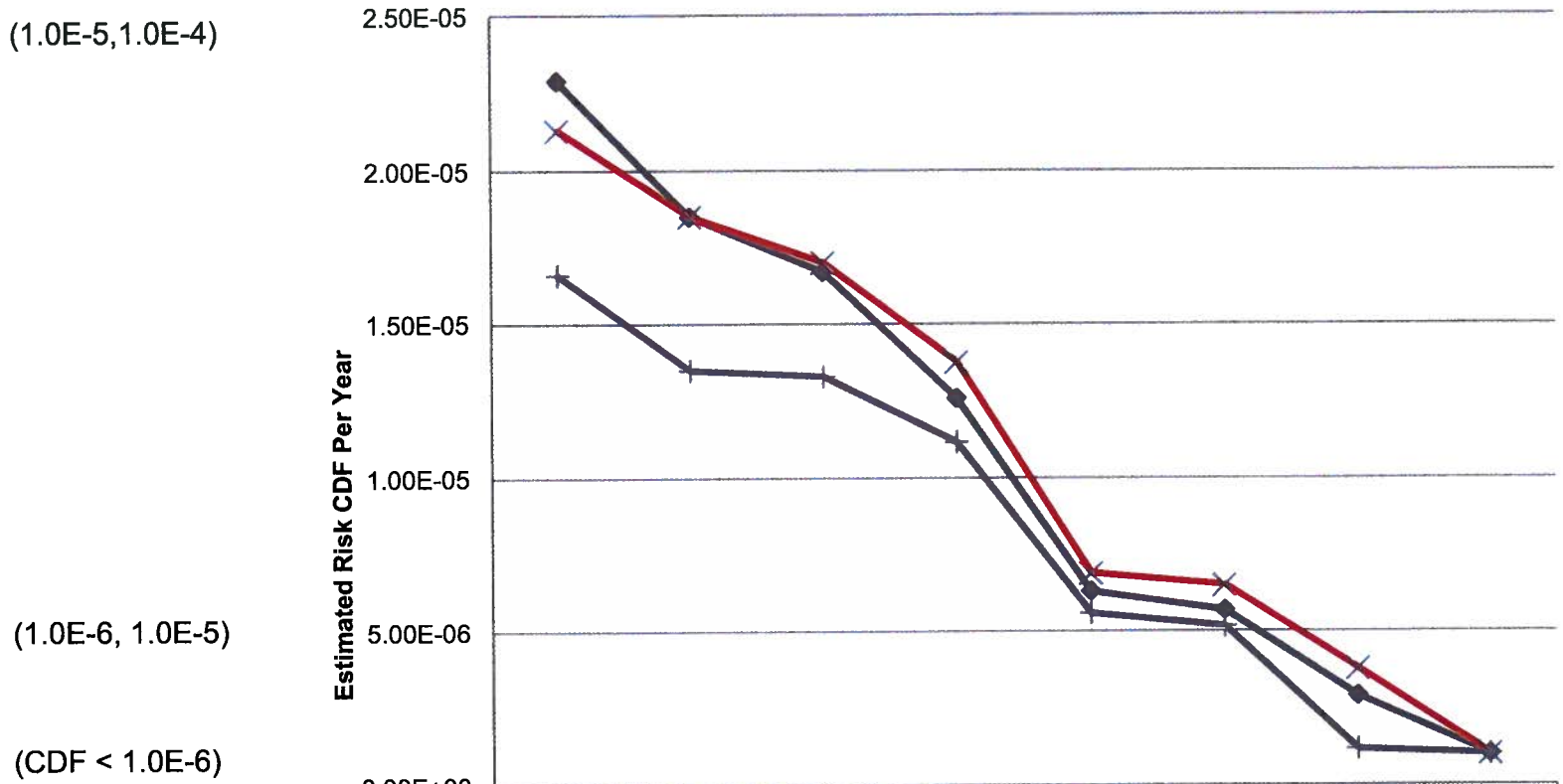
NRC Deputy Executive Director for Reactor and Preparedness Programs
NRC Deputy Executive Director for Materials, Waste, Research, State Tribal and Compliance Programs
NRC Director, Office of Enforcement
NRC Director, Office of Nuclear Reactor Regulation
NRC Senior Resident Inspector, Browns Ferry Nuclear Plant

ENCLOSURE 1

Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Fire SDP Risk Reduction Curve

Fire SDP Risk Reduction Curve (Current Schedule)



	SDP Original Value	Pull-Forward As Currently Scoped (Oct 2010)	Incipient Fire Detection SDB (Q1 2011)	Provide Divisional Tray Protection (Q1 2011)	SSI With Alternate Safe Shutdown Path (Q2 2011)	SSI Revision (Phase 1, FA 25, 26)	SSI Revision (Phase 2, FAs CDF > 1E-6)	NFPA 805 Based SSI (2012)
Extrapolated SDP Risk for Unit 1	2.29E-05	1.85E-05	1.67E-05	1.26E-05	6.30E-06	5.68E-06	2.89E-06	1.00E-06
Extrapolated SDP Risk for Unit 2	2.13E-05	1.85E-05	1.70E-05	1.38E-05	6.88E-06	6.49E-06	3.76E-06	1.00E-06
Extrapolated SDP Risk for Unit 3	1.66E-05	1.35E-05	1.33E-05	1.12E-05	5.58E-06	5.15E-06	1.16E-06	1.00E-06

ENCLOSURE 2

Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Final Significance Determination – Appeal
Cornerstone: Mitigating Systems

ENCLOSURE 2

Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Final Significance Determination – Appeal
Cornerstone: Mitigating Systems

Introduction

The NRC's Final Significance Determination (FSD) (Reference 1) characterized a performance deficiency associated with the Browns Ferry Nuclear Plant (BFN), Unit 1, low pressure coolant injection (LPCI)/residual heat removal (RHR) outboard inspection valve, 1-FCV-74-66, as a Red finding. The NRC considered the valve to be inoperable from March 13, 2009, to October 23, 2010, when the valve failed to open when operators attempted to place RHR Shutdown Cooling loop II in service to support Unit 1 refueling outage activities. As a result, the RHR loop II subsystem was inoperable while Unit 1 was operating in Mode 1.

Initially, in the NRC's preliminary greater than Green significance determination (Reference 2), the NRC described the performance deficiency associated with the valve as a failure to establish adequate design control and perform adequate maintenance on the valve. The preliminary determination stated that visual inspections had identified that the valve disc was stuck in the valve seat, blocking RHR loop II flow. The disc was separated from the valve disc skirt/stem assembly. Two fillet welds between the disc skirt and the disc were fractured. The original statement of the performance deficiency was based upon an assessment that the welds were undersized, a disc skirt locking key was missing, and a thrust washer between the stem and the disc was missing. The performance deficiency also focused on maintenance in 2006 in which 1-FCV-74-66 was refurbished and the valve stem was replaced.

At a Regulatory Conference on April 4, 2011, Tennessee Valley Authority (TVA) presented the results of its root cause evaluation based on comprehensive forensics performed on the valve. TVA determined that the disk separation from the disc skirt/stem assembly resulted from an original manufacturing defect – undersized threads in the disc/skirt assembly. The valve was purchased from a vendor as an assembly in 1968 under the vendor's 10 CFR Part 50, Appendix B Quality Assurance program. The valve was not taken apart to perform receipt inspections, since the manufacturer provided the required certifications and no reasonable basis existed thereafter to measure the threads and identify the undersized thread condition. The evaluation determined that undersized welds and the missing thrust washer and locking key were not the root cause or contributing causes of the valve failure. Accordingly, the root cause analysis determined that no licensee performance deficiency existed. The degraded condition was not the result of a failure to establish adequate design control or perform adequate maintenance as suggested in the NRC Report (Reference 2).

TVA also presented at the Regulatory Conference the results of detailed analyses and laboratory testing by Performance Improvement International (PII) to determine the valve's capability to function in its as-found condition (i.e., separation of disc from disc skirt/steam assembly). Based on an energy balance approach to determine the energy that would be supplied to the valve body and associated deflection from each stem stroke, and the vibration

effect and unseating force on the valve disc, PII determined that the disk would have released from the seat and provided proper flow within seven minutes. This was supported by evidence that the disc separated in November 2008, but nonetheless the valve loosened and operated as a check valve capable of allowing flow in March 2009. Flow within at most seven minutes would render the valve functional for the limiting Appendix R fire event in the significance determination risk assessment. PII stated that it had "very high" confidence in its findings.

In the FSD the NRC acknowledged that the weld configuration on 1-FCV-74-66 reasonably achieved its design function to prevent unthreading of the disc to skirt joint. Therefore, the NRC acknowledged the design aspects originally considered as part of the performance deficiency associated with the preliminary finding may not have been a primary contributor to the valve failure.

Nonetheless, the NRC in the FSD re-defined the performance deficiency related to design control and maintenance of the valve. The NRC stated that the required BFN's Inservice Test (IST) Program provided TVA with an opportunity to preclude and/or identify the LPCI valve failure sooner. Based on certain specific information requested by NRC and provided by TVA after the regulatory conference, the NRC concluded that certain aspects of the TVA IST program were deficient. In summary, the NRC FSD stated:

> Based on our review, the NRC has determined that certain aspects of TVA's IST program were inadequate. Namely, the NRC determined that TVA's failure to implement an IST program in accordance with the American Society of Mechanical Engineers (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addenda, 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 LPCI Outboard Injection Valve 1-FCV-74-66. The NRC has concluded that TVA's IST program inadequacy was well within its purview, and represents a performance deficiency.

TVA's appeal with respect to the FSD is based upon TVA's position that BFN's IST Program was not inadequate as detailed in this enclosure and that no licensee performance deficiency related to IST was involved in the valve failure. This issue was not addressed at the Regulatory Conference by TVA because NRC did not, prior to the conference, identify the IST program as an issue. The NRC's position in the FSD regarding BFN's IST Program is not based on a complete understanding of the issues since TVA did not have an opportunity to address the matter during the Regulatory Conference, and raises significant generic issues.

In accordance with Manual Chapter 0609, Attachment 0609.02, TVA maintains that an appeal of the FSD is appropriate because:

- the NRC's significance determination process (SDP) was inconsistent with the applicable guidance or lacked justification – because no performance deficiency was involved, and

- new information is available to address the revised performance deficiency related to the adequacy of BFN's IST Program. The information was not presented at the regulatory conference only because BFN's IST Program had not been raised by the NRC as a performance deficiency and was not discussed.

IST Program Adequacy

The valve in question, Unit 1 RHR loop (system) II 1-FCV-74-66, serves primarily to throttle RHR flow during shutdown cooling operations. During normal operation, this valve is in the fully open position. Its safety function is to remain in the fully open position in the event of an accident so that the LPCI pump can inject water into the reactor core. While the operators may use this valve to throttle LPCI flow post-accident, its primary safety function is to remain open. Furthermore, neither ultrasonic or radiographic testing can be used to determine the obturator position due to the valve configuration and construction materials.

TVA considers that BFN's IST Program is in full compliance with the American Society of Mechanical Engineers (ASME) Operation and Maintenance of Nuclear Power Plants Code (OM Code), 1995 Edition, 1996 Addenda, paragraphs ISTC 4.1 and ISTC 4.2.3, as mandated by NRC via 10 CFR 50.55a. The program as applied to this and similar valves is consistent with NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Revision 1. In addition, benchmarking determined that BFN's IST Program approach for implementing ISTC 4.1 and ISTC 4.2.3 is consistent with industry practices. These conclusions are in agreement with the professional opinion of industry-recognized experts as attested to in the attached affidavit (Attachment 2). The following discussion also supports this conclusion.

ISTC 4.1 covers testing performed on a 2-year basis. The first sentence of ISTC 4.1 states:

> *"Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated."*

ISTC 4.1 is thus very specific in that it:

a. Provides the scope - "valves with remote position indicators"
b. Mandates the method - "shall be observed locally"
c. Specifies frequency - "at least once every 2 years"
d. Defines a purpose - "verify that valve operation is accurately indicated"

This requirement must be taken in its entirety. It is improper to focus on the purpose without also focusing on the scope, method, and frequency. The following table shows how TVA complies with each aspect of this requirement through BFN's IST Program procedure 1-SR-3.3.3.1.4 (H II).

ISTC 4.1 - First Sentence	TVA BFN Procedure 1-SR-3.3.3.1.4(H II)
Provides scope - valves with remote position indicators	Section 1.2 states that the scope encompasses inservice testing of remote position indicators of RHR System II valves (including 1-FCV-74-66)
Mandates the method - shall be observed locally	Section 7.2, steps [4] through [9] verifies, control room lights and visually (locally) checks valve position
Specifies frequency - at least once every 2 years	Section 1.3 requires test performance every 2 years per ASME OM Code
Defines a purpose - verify that valve operation is accurately indicated	Section 1.1 states this procedure verifies that valve positions are accurately indicated

The remaining sentences of ISTC 4.1 are not requirements applicable to 1-FCV-74-66 as shown in the table below.

ISTC 4.1 - Remaining Sentences	Comments
Where practicable, this local observation should be supplemented by other indications such as use of flowmeters or other suitable instrumentation to verify obturator position.	While it is known that the NRC does not recognize official Code Interpretations, TVA considers that it is important to point out that ASME OM Code Interpretation 99-9 and Code Interpretation Question 1 of Inquiry Tracking Number 11-913 (Attachment 1) clarify that use of supplemental indications is not a requirement of the ASME OM Code. While TVA does not document use of supplemental indications to verify obturator position of 1-FCV-74-66, normal operating evolutions and IST full flow testing of the downstream check valve provide supplemental indication of 1-FCV-74-66 obturator position, although not specifically documented as such in the IST Program. A similar valve at BFN experienced disc-to-stem separation in 1974. The root cause was determined to be unthreading of the disc from the skirt/stem due to flow induced vibration. The design was changed for all similar valves at BFN following the 1974 event to eliminate the flow induced vibration failure mechanism. This corrective action was fully effective in that separation due to flow induced vibration did not recur over the subsequent 36 years. In 2008, degradation of various residual heat removal heat exchanger service water (RHR

	HX SW) outlet flow control valves was found, including disc to stem separation (see NRC Special Inspection Report 05000259, 260, and 296/2008007). This was due to flow-induced vibration from throttling, a different failure mechanism than the one associated with valve 1-FCV-74-66 in 2010. There was no other BFN or industry operating experience which indicated the current design of valve 1-FCV-74-66 was susceptible to disc-to-skirt/stem separation caused by a manufacturing defect of undersized threads leading to a pull-out. Therefore, there was no reason to consider formal documented use of supplemental methods of remote position indication. The 2010 stem-to-disc separation event was detected by normal outage operating evolution of placing RHR System II in shutdown cooling mode. Once the separation was detected, TVA took corrective action to disassemble and examine all similar valves across the three units for common cause as soon as conditions permitted. No other instances of incipient disc-to-skirt/stem failure were detected.
These observations need not be concurrent.	For the reasons just noted, IST remote position verification (local observation) is not performed concurrent with other supplemental indication of 1-FCV-74-66 obturator position which occurs during the course of system operating evolution and IST full flow testing of the downstream check valve. Supplemental indication of obturator position is not specifically documented as such in the IST program.
Where local observation is not possible, other indications shall be used for verification of valve operation.	Local observation of 1-FCV-74-66 is possible and performed in accordance with procedure 1-SR-3.3.3.1.4(H II). Therefore, this requirement is not applicable.

ISTC 4.2.3, which is also discussed in the FSD, covers testing that is performed on a quarterly frequency. It states:

> *The necessary valve obturator movement shall be determined by exercising the valve while observing an appropriate indicator, such as indicating lights that signal the required change of obturator position, or by observing other evidence,*

such as changes in system pressure, flow rate, level, or temperature, that reflects change of obturator position.

Again, this provision is very specific in that it:

a. Provides the test scope – "the necessary valve obturator movement shall be determined"
b. Mandates the method – "by exercising the valve while observing an appropriate indicator"
c. Provides examples of appropriate indicators, such as "indicator lights that signal the required change of obturator position," or "by observing other evidence, such as changes in system pressure, flow rate, level, or temperature, that reflects change of obturator position."

This requirement must be taken in its entirety. It is improper to focus on the scope without also focusing on the method and examples. The table below shows how TVA complies with each aspect of this requirement.

ISTC 4.2.3	TVA BFN Procedure 1-SI-3.6.1.3.5 (RHR II)
Provides test scope - the necessary valve obturator movement shall be determined	Section 1.1 states the purpose is to verify the opening time of 1-FCV-74-66. In addition, Section 1.2 states the valve will be operated through at least one complete cycle and will measure the time to stroke the valve to the position required to perform its safety function.
Mandates the method - by exercising the valve while observing an appropriate indicator	Section 7.2, steps [1] through [4] exercises and measures the valve stroke time while observing indicating lights
Provides examples of appropriate indicators, such as:	
• indicator lights that signal the required change of obturator position,	Section 3.0B defines the stroke time as the period from initial switch movement to panel indication of completed valve travel. Section 7.2, steps [1] through [4] exercises and measures the valve stroke time using indicating lights. Indicating lights are specifically cited as an example of an appropriate indicator to determine necessary obturator movement.
• or by, observing other evidence, such as changes in system pressure, flow rate, level, or temperature, that reflects change of obturator position.	This option was not used because the other example cited as an appropriate indicator (indicating lights) is used. The use of the "or by" in the Code language indicates that only one method is required.

	In addition, OM Code Interpretation Question 2 of Inquiry Tracking Number 11-913 (Attachment 1) clarifies that when indicator lights are used, it is not a Code requirement to also observe other evidence that reflects change of obturator position. Furthermore, Section 1.3 states the test frequency as once per 92 days. Exercise testing is typically performed when the portion of the system, in which the valve is located is in static conditions. Specifically, valve 1-FCV-74-66 is located in a standby system so there is no possibility of a change in flow, pressure, temperature, etc., during the quarterly (i.e., 92 day) exercise test performed to satisfy ISTC 4.2.3.

NUREG-1482, Revision 1, Section 4.2.7 provides NRC guidance and recommendations regarding remote position indication verification of valves. Revision 1 of NUREG-1482 is the version applicable to BFN. NUREG-1482, Revision 1, specifically states that Revision 0 (1995) is only valid for licensees who have not updated their IST programs to the 1995 Edition/1996 Addenda or a later edition of the Code. BFN's IST Program complies with the 1995 Edition/1996 Addenda as required by 10 CFR 50.55a(b)(3) and 10 CFR 50.55a(f)(4)(ii). NUREG-1482, Revision 1, specifically states that it is applicable to the 1995 Edition/1996 Addenda and later editions and addenda of the Code. The following table compares this guidance with TVA's implementation through testing of valve 1-FCV-74-66.

NUREG-1482, Revision 1, Section 4.2.7, Verification of Remote Position Indication for Valves by Methods Other Than Direct Observation	**Comments**
Introduction: OM Subsection ISTC 3700 [4.1] requires that valves with remote position indicators must be observed at least once every 2 years to verify that valve position is accurately indicated. • Many valves (such as sealed solenoid valves and valves with enclosed stems) have no provision for verifying the position by direct observation. • To verify the position by observation, licensees can disassemble the valve, which could introduce additional valve failure mechanisms. • Other methods (such as nonintrusive techniques, causing the flow to begin or cease, leak testing, and pressure	The purpose of Section 4.2.7 is to address those valves which cannot be observed locally. BFN valve 1-FCV-74-66 can be observed locally.

testing) can yield a positive indication of position.	
NRC Recommendation: If licensees cannot verify remote valve position by local observation at the valve, an acceptable approach is for the licensee to observe operational parameters (such as leakage, pressure, and flow) that give a positive indication of the valve's actual position(s). • This is consistent with Subsection ISTC 3700 [4.1]. • The staff determined that the use of this portion of the OM Code would not require relief if the licensee implements all requirements of Subsection ISTC 3700 [4.1]. • No other related requirements apply. However, Commission approval is still required pursuant to 10 CFR 50.55a(f)(4)(iv).	This recommendation is not applicable to BFN valve 1-FCV-74-66 because it can be observed locally.
For certain types of valves that can be observed locally, but for which valve stem travel does not ensure that the stem is attached to the disk, the local observation should be supplemented by observing an operating parameter as required by Subsections ISTC 3700 and 3520 [4.1, 4.2, and 4.5].	The use of "should" in this recommendation is in agreement with ISTC 4.1, as further clarified by ASME Code Interpretation 99-9 and Code Interpretation Question 1 of Inquiry Tracking Number 11-913 (Attachment 1) which state that use of supplemental indications is not a requirement of the ASME OM Code. As noted in comments to ISTC 4.1 above, TVA did not have information or reason to suspect the current design valve 1-FCV-74-66 was susceptible to disc-to-skirt/stem separation, and so there was no compelling reason to consider use of supplemental methods of remote position indication verification.
Basis for Recommendation: Accurate position indication for safety-related valves is important for reactor operation under all plant conditions.	No Comment
The Code requires licensees to verify the accuracy of the remote position indication for all valves in the IST Program that have remote position indication.	No Comment

Subsection ISTC 3700 [4.1] states that where local observation is not possible, licensees shall use other indications to verify operation.	This is not applicable to BFN valve 1-FCV-74-66 because local observation is possible.
Such indications are also useful to ensure that a valve disk is connected to the stem. Licensees are not able to verify the accuracy of remote position indication by local observation of many valves, such as those with enclosed stems or sealed solenoid valves, where the valves do not have position indicators, such as pointers, on the valve actuators. Many positive means are available to verify the indication that a valve is open or closed. For example, leak rate testing may yield positive indication that the disk is in the closed position. In addition, an in-line flow rate instrument can indicate system flow or flow stoppage. System pressures or differential pressure across a valve seat may also give a positive indication of actual valve position.	No Comment This is not applicable to BFN valve 1-FCV-74-66 because local observation is possible. TVA does not document use of supplemental indications to verify obturator position of 1-FCV-74-66. Normal operating evolutions and IST of downstream check valve provide supplemental indication of 1-FCV-74-66 position indication. The 2010 stem-to-disc separation was detected by normal outage operating evolution of placing RHR System II in shutdown cooling mode. Once the separation was detected, TVA took immediate action to disassemble and examine all similar valves across the three units for common cause. No other instances of disc-to-skirt/stem degradation were detected.

Based on the above analysis, TVA considers that the BFN IST program complies with the applicable requirements of the OM Code and is consistent with NRC guidance in NUREG-1482, Revision 1. This conclusion is consistent with the professional opinion of industry experts as attested to in the attached affidavit (Attachment 2). In this regard, some of the conclusions noted in the FSD letter appear to be based on an incomplete understanding of BFN's IST Program details. These points are addressed in the following table. (References are to Enclosure 2 of the FSD letter.)

Enclosure 2, page 3, first full paragraph	**Comments**
The NRC identified that the required IST program provided TVA with an opportunity to preclude and/or identify the LPCI valve failure sooner.	IST-related remote position indication testing is performed on a 2-year frequency. The actual detection method for the 1-FCV-74-66 degraded condition was a normal system operation evolution. This evolution occurred before there would have been an opportunity to perform IST remote position indication test using supplemental methods such as use of

	flowmeters, etc. The system must be placed in service before an IST test can be performed using flowmeters. Note that the previous supplemental test (full flow check valve test) was successfully performed two years earlier.
In this regard, the post-conference supplemental information provided by TVA highlighted that certain aspects of its IST program were deficient	TVA stated, in the post-conference supplemental information provided to NRC, that BFN's IST Program is in compliance with the OM Code and the guidance provided in NUREG-1482, Revision 1. TVA further clarified that although it is not a Code requirement to use supplemental methods such as flowmeters in addition to local observation for remote position indication verification, TVA performs an IST-related full flow check valve test which can be credited as a supplemental method. In this case, the full flow check valve test is performed on a cold shutdown frequency, but does not specifically take credit for remote position indication verification testing of 1-FCV-74-66.
Namely, TVA's failure to implement an IST Program in accordance with the requirements of ASME 1995 Edition, 1996 Addenda, 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 LPCI Outboard Injection Valve 1-FCV-74-66	TVA fully complies with ISTC 4.1 as shown in Item 2 above. IST-related remote position indication verification testing is required to be performed at least once every 2 years in accordance with OM Code paragraph ISTC 4.1. BFN procedure 1-SI-3.2.21(II), Cold Shutdown Testing of 1-CKV-74-68, exercises check valve 1-CKV-74-68 to the full open position by verifying it is capable of passing at least 9,000 gpm Shutdown Cooling flow. This procedure uses valve 1-FCV-74-66 to throttle Shutdown Cooling flow as needed to achieve a flow rate of at least 9,000 gpm. The November 14, 2008 performance of this procedure shows that Section 7.2, Step [1.3] recorded an initial Shutdown Cooling flow of 7,000 gpm as indicated on 1-FI-74-64. Then, Section 7.0, Step [1.3] throttled 1-FCV-74-66 to obtain a minimum flow rate of 9,000 gpm as indicated on 1-FI-74-64. This step demonstrates that 1-FCV-74-66 was capable of throttling Shutdown Cooling flow and that 1-FCV-74-66 as well as 1-CKV-74-68 were both capable of passing the required flow rate of at least 9,000 gpm. This procedure does not specifically take credit for verifying obturator position of 1-FCV-74-66, but the actions are performed just the same as if IST credit was being taken.

	Degradation of 1-FCV-74-66 was detected when Shutdown Cooling was being placed in service in October 2010. This occurred before the opportunity to perform the next scheduled 1-SI-3.2.21(II).

Enclosure 2, page 3, second full paragraph	Comments
The in-service testing (IST) program implemented by TVA at Browns Ferry to meet the requirements of 10 CFR 50.55a, [incorporates by reference, with conditions, the American Society of Mechanical Engineers (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addenda], was inadequate to identify the inability of LPCI Valve 1- FCV-74-66 to perform its intended safety function.	BFN's IST Program complies with the OM Code Sections 4.1 and 4.2.3. The OM Code is designed to provide reasonable assurance of valve operational readiness. The Code requirements do not provide absolute assurance of valve operability. It is possible that some types of valve degradation may go undetected by inservice testing which fully complies with all Code requirements and recommendations. This was the case with valve 1-FCV-74-66.
Despite the operating experience with LPCI Valve 1-FCV-74-66 and other similar valves at Browns Ferry, Units 1, 2, and 3, the TVA program at Browns Ferry implemented in response to 10 CFR 50.55a(b)(3)(ii) did not ensure that LPCI Valve 1-FCV-74-66 continued to be capable of performing its design-basis safety function.	As explained above, a similar valve at BFN experienced disc-to-stem separation in 1974. The root cause was determined to be unthreading of the disc from the skirt/stem due to flow induced vibration. The design was changed to eliminate the flow induced vibration failure mechanism for all similar valves at BFN following the 1974 event. This corrective action was fully effective in that separation due to flow induced vibration has not recurred over the next 36 years. In 2008, degradation of various residual heat removal heat exchanger service water (RHR HX SW) outlet flow control valves was found, including disc to stem separation. This was due to flow-induced vibration from throttling, a different failure mechanism than the one associated with valve 1-FCV-74-66 in 2010. There was no other BFN or industry operating experience which indicated the current design of valve 1-FCV-74-66 was susceptible to disc-to-skirt/stem separation caused by a manufacturing defect of undersized threads leading to disc-to-skirt/stem pull-out.

Enclosure 2, page 3, last paragraph	Comments
In NUREG-1482 (April 1995), Guidelines for Inservice Testing at Nuclear Power Plants, the NRC staff discussed its interpretation of the ASME OM Code requirement for verification of remote position indication. . .	This is an incorrect reference. NUREG-1482, Revision 0 (April 1995) does not apply to BFN. NUREG 1482, Revision 1 specifically states Revision 0 is only valid for licensees who have not updated their IST programs to 1995 Edition/1996 Addenda or a later edition of the Code. BFN's IST Program complies with the 1995 Edition/1996 Addenda as required by 10 CFR 50.55a(b)(3) and 10 CFR 50.55a(f)(4)(ii). NUREG-1482, Revision 1, specifically states it is applicable to 1995 Edition/1996 Addenda and later editions and addenda.

Enclosure 2, page 4, first full paragraph	Comments
In Revision 1 to NUREG-1482, the NRC staff in Section 4.2.7 discussed its interpretation of the ASME OM Code for position indication verification. The staff continued to specify that the first sentence in ISTC 4.1 provides the requirement for verification of the accuracy of the remote position indication for all valves in the IST program with remote position indication. The staff noted that when a licensee cannot verify remote valve position by local observation, an acceptable approach is to observe operational parameters (such as leakage, pressure, and flow) that give a positive indication of the valve's actual position(s). The staff modified the discussion regarding supplementing local observations (i.e., "must" to "should") to reflect that the NUREG provided guidance for licensee IST programs.	NUREG-1482, Revision 1 revised the NUREG-1482, Rev. 0, Section 4.2.5 (equivalent to 4.2.7) language from "must" to "should." The term "must" is used numerous times in NUREG-1482, Revision 1 to denote requirements. Therefore, it seems clear the change from "must" to "should" in Section 4.2.7 was for a reason other than to reflect that the NUREG provides guidance for licensee IST Programs. The reasonable explanation for the change was because there is no Code language or other regulatory requirement which supports the use of "must."

Enclosure 2, page 4, second full paragraph	Comments
The acceptance criteria requires that valve position be "visually checked" in the correct position.	Procedure 1-SR-3.3.3.1.4(H II), Section 6.0B states, "Visual verification of valve position (open/closed) is accurately indicated by its remote position indicators (red, green lights) at the Main Control Room panels. Local valve position verification can make use of limit switches, position indicators on the valve, valve

	stem travel, etc., to determine valve position." Section 7.2, steps [4] through [9] check the control room indicating lights and visually check valve position locally.
This method of verification does not ensure that the disc is attached to the disc and therefore is unable to adequately verify valve internal operations, as described in the ASME code as obturator position.	There is an error in the enclosure where it states, ". . .disc is attached to the disc. . ." It should state, ". . .disc is attached to the stem. . ." This statement incorrectly implies it is a requirement of ISTC 4.1 that local observation of valve operation be supplement by other indications to verify obturator position. OM Code Interpretation 99-9 and Code Interpretation Question 1 of Inquiry Tracking Number 11-913 (Attachment 1) clarify this is not a requirement of the Code.
In addition, TVA had knowledge that this valve design is susceptible to stem and disc separation based on its design configuration.	TVA does not agree with this statement. A similar valve at BFN experienced disc-to-stem separation in 1974. The root cause was determined to be unthreading of the disc from the skirt/stem due to flow induced vibration. The design was changed to eliminate the flow induced vibration failure mechanism for all similar valves at BFN following the 1974 event. This corrective action was fully effective in that separation due to flow induced vibration did not recur over the subsequent 36 years. In 2008, degradation of various residual heat removal heat exchanger service water (RHR HX SW) outlet flow control valves was found, including disc-to-stem separation (see NRC Special Inspection Report 05000259, 260, and 296/208007). This was due to flow-induced vibration from throttling, a different failure mechanism than the one associated with valve 1-FCV-74-66 in 2010. There was no other BFN or industry operating experience which indicated the current design of valve 1-FCV-74-66 was susceptible to disc-to-skirt/stem separation caused by a manufacturing defect of undersized threads leading to a pull-out.

Enclosure 2, page 4, last paragraph	Comments
TVA testing acceptance criteria for implementing procedure, 1-SR 3.6.1.3.5, intended to meet the ASME code requirements via verification of the open/shut indicating lights during valve stroke and time testing.	As explained above, procedure 1-SR-3.6.5 (RHR II) complies with ISTC 4.2.3. Use of indicating lights is specifically cited in ISTC 4.2.3 as an example of an appropriate indicator to determine necessary obturator movement.
This surveillance was also unable to adequately verify obturator movement, and did not identify the stem and disc separation.	Procedure 1-SR-3.6.5 (RHR II) performs exercising and stroke time measurement of 1-FCV-74-66 on a quarterly frequency. The use of indicating lights is the only practical and appropriate indicator to determine obturator movement during plant modes of operation present during quarterly testing. Specifically, valve 1-FCV-74-66 is located in a standby system so there is no possibility of a change in flow, pressure, temperature, etc., during the quarterly (92 day) exercise test performed to satisfy ISTC 4.2.3. This approach is consistent with industry practices as attested to in the attached affidavit (Attachment 2).

Enclosure 2, page 5, first paragraph	Comments
The staff concluded that the opportunity existed for the testing program to discover the failure had the testing program met the code requirements to verify that valve operation is accurately indicated.	As discussed above, the IST program complies with Code requirements, as mandated by NRC via 10 CFR 50.55a, which are written to provide reasonable assurance of valve operational readiness. The 10 CFR 50.55a regulation and the Code requirements do not provide absolute assurance of valve operability. It is possible that some types of valve degradation may go undetected by inservice testing which fully complies with all Code requirements and recommendations. This was the case with valve 1-FCV-74-66.

Enclosure 2, page 5, second paragraph, first bullet	Comments
The staff questioned the licensee specifically on other actions beyond using the indication lights to verify obturator movement. The licensee responded that they use lights and local position indication (stem movement), because that meets the code requirements and	As discussed above, BFN's IST Program complies with ISTC 4.1 and ISTC 4.2.3. Furthermore, the BFN methods are supported by OM Code Interpretation 99-9 and Code Interpretation Question 1 of Inquiry Tracking Number 11-913 (attached) which have been

is consistent with industry practices. The staff has concluded that this was an improper interpretation and implementation of the code requirements.	approved through ASME which is the only organization authorized to interpret the OM Code.

Enclosure 2, page 5, second paragraph, second bullet	Comments
The staff additionally verified, with the licensee, that flow testing or flow parameter verification is not used to verify stem to disc separation. The staff did not find that the other licensee's procedures and actions provided a means to identify stem to disc separation. Specifically the staff considered flow testing on the 1-FCV-74-68 in the RHR loop II system. This procedure was not designed in any way for verification of 1-FCV-74-66 proper obturator movement and therefore was not able to meet ASME code requirements and continued ability of 1-FCV-74-66 to fulfill its safety function.	As performed in November 2008, procedure 1-SI-3.2.21(II) demonstrated full flow (9,000 gpm) was passed through check valve 1-CKV-74-68. This check valve is located downstream of valve 1-FCV-74-66. Therefore, 1-FCV-74-66 passed at least 9,000 gpm in order to meet the acceptance criteria for the full flow test of 1-CKV-74-68. Specifically, performance of this procedure used 1-FCV-74-66 to throttle shutdown cooling flow through 1-CKV-74-68 from 7,000 gpm to 9,000 gpm. Although, procedure 1-SI-3.3.21(II) does not specifically address position indication verification of 1-FCV-74-66, it is incorrect to conclude that this procedure would not have verified proper obturator movement. This procedure was not able to be performed in October 2010 because it requires RHR System loop II to be in service in shutdown cooling mode of operation. Degradation of 1-FCV-74-66 was detected when placing RHR System II into service for shutdown cooling.

Enclosure 2, page 5, third paragraph	Comments
Furthermore, TVA had operating experience that indicated this valve design is susceptible to separation failures.	A similar valve at BFN experienced disc-to-stem separation in 1974. The root cause was determined to be unthreading of the disc from the skirt/stem due to flow induced vibration. The design was changed to eliminate the flow induced vibration failure mechanism for all similar valves at BFN following the 1974 event. This corrective action was fully effective in that separation due to flow induced vibration did not recur over the subsequent 36 years.
Previous issues experienced on multiple valves (including 1-FCV-74-66) at Browns Ferry established that this valve was susceptible to stem and disc separation. One similar instance	In 2008, degradation of various residual heat removal heat exchanger service water (RHR HX SW) outlet flow control valves was found, including disc to stem separation. This was

resulted in a valve disc separating from the stem and becoming stuck in its seat. The previous failure was identified following a failure of the system to operate when being placed in service. Other valves at the Browns Ferry site also have experienced stem and disc separation issues (see NRC Special Inspection Report 05000259, 260, and 296/2008007, Section 4OA2.f).	due to flow-induced vibration from throttling, a different failure mechanism than the one associated with valve 1-FCV-74-66 in 2010. There was no other BFN or industry operating experience which indicated the current design of valve 1-FCV-74-66 was susceptible to disc-to-skirt/stem separation caused by a manufacturing defect of undersized threads leading to a pull-out. Review of NRC Information Notices, inspection reports, and other communications did not identify any instances where deficiencies in implementation of ASME OM Code IST Programs were cited in relation to disc-to-stem failures. Therefore, no precedent existed which would have warranted TVA corrective action.

Enclosure 2, page 5, fourth paragraph	**Comments**
Based on our review, the NRC has determined that certain aspects of TVA's IST Program were inadequate. Namely, the NRC determined that TVA's failure to implement an IST Program in accordance with the American Society of Mechanical Engineers (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addenda, 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 LPCI Outboard Injection Valve 1-FCV-74-66.	As discussed above, the IST program complies with Code requirements, as mandated by NRC via 10 CFR 50.55a. TVA's IST Program was submitted to the NRC in 2002 as required by OM Code ISTA 1.4(c). Benchmarking determined that BFN's IST Program approach for implementing ISTC 4.1 and ISTC 4.2.3 is consistent with industry practices as attested to in the attached affidavit (Attachment 2). In addition, the following facts demonstrate that IST would not have provided a more timely identification of the 1-FCV-74-66 degradation. • IST-related remote position indication testing is performed on a 2-year frequency. • The actual detection method for the 1-FCV-74-66 degraded condition was a normal system operation evolution. • This evolution occurred before there would have been an opportunity to perform IST remote position indication test using supplemental methods such as use of flowmeters, etc. • The system must be placed in service before an IST test can be performed using flowmeters.

	Review of NRC Information Notices, inspection reports, and other communications did not identify any instances where deficiencies in implementation of ASME OM Code IST Programs were cited in relation to disc-to-stem failures. Therefore, no precedent existed which would have warranted TVA taking corrective action.
The NRC has concluded that TVA's IST Program inadequacy was well within its purview, and represents a performance deficiency	TVA has shown BFN's IST Program to be in accordance with Code requirements and industry practices. Therefore, it is not reasonable to consider this a performance deficiency simply because valve degradation was detected by a means other than the IST program.

Summary

BFN's IST Program meets applicable requirements and guidance documents and, therefore, testing of valve 1-FCV-74-66 does not represent a performance deficiency. It is in full compliance with the American Society of Mechanical Engineers (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addenda, paragraphs ISTC 4.1 and ISTC 4.2.3, as mandated by NRC via 10 CFR 50.55a.

Benchmarking determined that BFN's IST Program approach for implementing ISTC 4.1 and ISTC 4.2.3 is consistent with industry practices. For example, BFN's sister plants have the same valves in the same application, and implement the requirements of ISTC 4.1 and 4.2.3 in the same manner as BFN.

Review of NRC Information Notices, inspection reports, and other communications did not identify any instances where deficiencies in implementation of ASME OM Code IST Programs were cited in relation to disc-to-stem failures. Therefore, no precedent existed which would have warranted TVA taking corrective action.

The ASME OM Code language and its predecessor requirements related to remote position indication verification and valve exercising (ISTC 4.1 and ISTC 4.2.3) have remained unchanged from 1987 through the OM Code edition/addenda currently endorsed by NRC. Current Code IST requirements are not designed such that they are capable of detecting degradation of the disc-to-stem connection. They are only capable of detecting failure after it has occurred. The NRC interpretation of the applicable OM Code requirements reflected in the FSD letter would have a significant impact on valve testing at BFN and other plants throughout the industry.

BFN's IST Program would not have provided a more timely identification of the 1-FCV-74-66 degraded condition for the following reasons.

- IST-related remote position indication testing is performed on a 2-year frequency.

- The actual detection method for the 1-FCV-74-66 degraded condition was a normal system operation evolution.

- This evolution occurred before there would have been an opportunity to perform IST remote position indication test using supplemental methods such as use of flowmeters, etc.

- The system must be placed in service before an IST test can be performed using flowmeters.

Previous valve disc to stem separations at BFN were due to flow-induced vibration and were adequately addressed. Therefore, TVA did not have reason to consider use of supplemental methods in addition to local observation for IST remote position indication verification.

It is possible to detect degradation of a valve through normal plant operating evolutions. Therefore, detection of valve degradation by a means other than the IST program does not necessarily represent an IST program inadequacy.

BFN's IST Program does not represent a performance deficiency. Rather BFN's IST Program is in compliance with OM Code, ISTC 4.1 and ISTC 4.2.3 and industry practices as attested to in the affidavits from industry-recognized experts provided in Attachment 2 to this enclosure.

No Performance Deficiency

NRC Manual Chapter 0609, "Significance Determination Process," Section 0609-08, paragraph 08.01, addresses the development of inspection findings to be considered in the Reactor Oversight Program and states (emphasis in original): "Findings must represent a deficiency in a licensee's performance." TVA has previously demonstrated that the root cause of the inoperable valve 1-FCV-74-66 was an original manufacturing defect that occurred prior to initial operation of BFN. The original manufacturer's design requirements were not met, resulting in undersized disc skirt threads at the disc skirt/stem assembly connection. No receipt inspection of the thread size of this nature and classification was required. The manufacturer applied an Appendix B QA program and provided certification documentation.

TVA also reviewed the maintenance history for the valve and determined that the disc skirt/stem assembly of the original assembly was not replaced prior to the failure and no work was performed that required measuring/confirming the disc skirt thread size. Moreover, as discussed above, the IST program at BFN complied with the applicable requirements of Section ISTC 4.1 of the OM Code. Based on industry benchmarking and expert assessments, the IST program at BFN was consistent with NRC requirements and industry practices, and did not reasonably provide an opportunity to identify the disc separation. Furthermore, based on the information provided in the previous section, TVA has concluded that there were no methods other than the ones implemented by BFN's IST Program to detect the disc and disc skirt/stem assembly separation experienced by valve 1-FCV-74-66. Given that it was not reasonably within TVA's ability to foresee, identify, or prevent the valve failure, there was no licensee performance deficiency to support a Red Finding or to warrant the related regulatory response based on the Reactor Oversight Process Action Matrix.

This conclusion is also consistent with the NRC's Enforcement Policy. TVA recognizes that licensees are ordinarily responsible for the acts of their vendors and contractors. However, Section 3.5 of the Policy provides that "the NRC may refrain from issuing enforcement actions for violations resulting from matters not within a licensee's control, such as equipment failures that were not avoidable by reasonable licensee QA measures or management controls." This policy is also reflected in NRC Enforcement Manual, Section 5.1.4.C, which allows NRC discretion not to issue enforcement actions where the licensee's root cause analysis demonstrates that the equipment failure could not have been avoided or detected. NRC Manual Chapter 06112, page 18, also covers enforcement discretion where there is no performance deficiency.

TVA is aware of several cases in which the NRC has previously exercised this discretion under the Reactor Oversight Program — even for matters that may have involved regulatory violations and greater than Green findings. These examples are:

- Braidwood Station, Units 1 and 2, NRC Integrated Inspection Report and Exercise of Enforcement Discretion 05000456/2006005; 05000457/2006005 and 05000456/2006013; 05000457/2006013, EA-06-225, dated February 13, 2007: equipment failure could not have been avoided or detected by quality assurance program or other related control measures because Type 316 stainless steel heater sleeve material was not known to be susceptible to cracking.

- Indian Point Nuclear Generating Station, Unit 3, NRC Integrated Inspection Report 05000286/2008005 and Exercise of Enforcement Discretion, dated February 5, 2009: safety injection pump breaker did not close during routine operation, but (although a violation) the breaker subcomponent problem that resulted in the inoperability was not within the licensee's ability to foresee and correct and, as a result, the NRC did not identify a performance deficiency.

- Three Mile Island Station, Unit 1, NRC Integrated Inspection Report 5000289/2009002, Exercise of Enforcement Discretion, dated May 13, 2009: a failed electrical relay caused a decay heat river water pump to be inoperable in excess of allowed outage time, but (although a violation) was not the result of a performance deficiency.

A similar outcome is particularly warranted in this case. The root cause for the equipment failure relates to manufacturing by a vendor prior to initial BFN Unit 1 start up. This does not reflect current licensee performance and thus should not trigger a current, enhanced regulatory response through the Reactor Oversight Process. Also, while TVA does not agree that BFN's IST Program was insufficient, the specific issues related to testing of valves of this type are being addressed through a review of BFN's IST Program by a team of industry-recognized IST experts, including the principal author of NUREG-1482, Revision 1, for compliance or implementation issues. Any generic issues related coming out of this review related to BFN's IST Program will be addressed prospectively on a generic basis, following appropriate industry interactions.

Action Matrix Deviation

If, notwithstanding the lack of any TVA performance deficiency, the NRC chooses to uphold the FSD, TVA requests that the NRC approve a deviation from the Action Matrix in accordance with Manual Chapter 0305, Section 12.06. Deviations are to be utilized in "rare instances," such as the present one, in which the regulatory actions dictated by the Action Matrix are not appropriate. The FSD, when entered into the Action Matrix, would lead to an inappropriate level of regulatory attention and diversion of licensee resources and management focus. There are several bases for the conclusion that a deviation is appropriate.

First, as discussed above, the FSD does not reflect on current TVA performance. The valve failure was caused by an original manufacturing defect. And, even if the NRC maintains that BFN's IST Program was a missed opportunity to identify disc separation in the valve, that conclusion would be based on new regulatory expectations for IST programs derived from this operating experience. These expectations were not anticipated in the industry or by TVA, and therefore cannot be reasonably considered to reflect on TVA's overall, current performance.

Second, as discussed at the Regulatory Conference, the risk significance for the FSD is driven by a limiting Appendix R fire event. TVA has previously initiated actions to address fire safe shutdown strategies at BFN and reduce plant risk due to serious fire events, including by transition to a NFPA 805 fire protection methodology (see the Fire SDP Reduction Curve in Enclosure 1). Specifically, the Safe Shutdown Instructions (SSIs) that would be used in the event of an Appendix R fire have been revised as described during the Regulatory Conference. This revision provides the operators with guidance to use systems other than the ones specified in the SSIs to provide cooling water to the reactor should the systems specified in the SSIs not function as expected. All of these actions will address the fire risk issues that would be addressed in a regulatory response under the Action Matrix.

Similarly, fire risk issues at BFN were addressed in connection with previous Yellow and White Findings under the Reactor Oversight Process issued on April 19, 2010 (EA-09-307). The NRC conducted inspections in accordance with Inspection Procedure 95002, and the greater than Green findings have since been closed. However, the present finding — with risk dominated by the same fire risk considerations — would drive a second assessment potentially including many of the same issues. NRC Manual Chapter 0308, Attachment 3, page 9, provides that there will be multiple inputs to the Action Matrix for conditions that overlap in time only if the findings are "separate and independent" of each other. In the present circumstances, the risk involved in each of the past and current findings is driven by the same current fire safe shutdown strategies at BFN, which are already in the process of being revised. The findings are, therefore, not separate and independent and should not be double-counted in the Action Matrix.

For all of these reasons, TVA considers that a deviation from the Action Matrix would be appropriate if the FSD is upheld.

Clarifications

The following clarifications are provided to address statement in the NRC's May 9, 2011 FSD letter.

- Cover letter, ninth paragraph, states that the discussion of BFN's IST Program issue during the April 29, 2011 conference call between the NRC and TVA staffs included ". . . an exchange of information regarding the implementation of the OM Code requirements at Brown Ferry . . ." In fact, the TVA staff on that conference call was only able to state that they would investigate the issue, and ensure that it was entered into the Corrective Action Program since that was the first time this issue was identified to them.

- Enclosure 2, Section 2, "Valve Functionality, NRC Response," states in part that TVA's determination of the maximum time it takes for a disc to be freed from its seat was based on the ". . . results of a single test. . ." This same statement is made later regarding testing to determine the minimum time it takes for a disc to be freed from its seat. In fact there were five tests performed to determine the range of times it would take, from minimum to maximum, for a disc to be freed from its seat. This information was provided to the NRC in response to Round 2 Question No. 3 on April 13, 2011.

- Enclosure 2, Section 2, "Valve Functionality, NRC Response," states in part that . . . "uncertainties in the testing methodology which challenged the level of confidence that the testing was representative of actual system conditions. . . ." In fact, testing uncertainties and the confidence level that the results reflected actual valve performance were also discussed in response to Round 2 Question No. 3 on April 13, 2011. The confidence level was specified in that answer as 99.7%.

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June 7, 2011

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Subject: ISTC-4.1 (ASME OM Code -1995 through OMa -1996 Addenda) and ISTC-3700 (ASME OM Code-1998 through 2004 Edition with 2006 Addenda); and

ISTC-4.2.3 (ASME OM Code -1995 through OMa - 1996 Addenda) and ISTC-3530 (ASME OM Code-1998 through 2004 Edition with 2006 Addenda).

Reference: Your Email Dated May 24, 2011
Item: Inquiry Tracking Number 11-913

Question (1): If it is practicable, is it a requirement of ISTC-4.1 (ISTC-3700) that local observation of valve operation be supplemented by other indications to verify obturator position?

Reply (1): No

Question (2): If remote indicating lights provide confirmation of changes in obturator position, is it a requirement of ISTC-4.2.3 (ISTC-3530) to also observe other evidence, such as changes in system pressure, flow rate, level, or temperature, that reflects change of obturator position?

Reply (2): No

Sincerely,

Robert Horvath

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Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Final Significance Determination — Appeal
Cornerstone: Mitigating Systems

JOINT AFFIDAVIT OF STEVEN UNIKEWICZ,
WESLEY ROWLEY, AND MARK GOWIN IN SUPPORT OF FSD APPEAL

Steven M. Unikewicz

1. [SMU] My name is Steven M. Unikewicz. I am currently employed as Principal Engineer, Program Manager, for Alion Science and Technology in Vienna, Virginia. In my position at Alion Science and Technology I am responsible for providing component and plant system evaluations and analysis to the nuclear industry.

2. [SMU] Prior to joining Alion Science and Technology, I was a Senior Mechanical Engineer, Mechanical Branch, Office of Nuclear Reactor Regulation, at the U.S. Nuclear Regulatory Commission involved in the review and development of codes, standards and regulations regarding the inservice testing of components at nuclear power generating stations.

3. [SMU] While at the NRC, I prepared NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants." NUREG-1482, Revision 1, provides licensees guidelines and recommendations for developing and implementing programs for the inservice testing of pumps and valves at commercial nuclear power plants.

4. [SMU] This affidavit is prepared to support Enclosure 2 to the "Final Significance Determination — Appeal Cornerstone: Mitigating Systems" for Browns Ferry

Nuclear Plant, Unit 1 ("FSD Appeal"). A statement of my professional qualifications is attached to this affidavit as Attachment A.

5. [SMU] With respect to this joint affidavit, I have included my initials within brackets (*i.e.*, [SMU]) immediately preceding those sections I prepared.

C. Wesley Rowley

6. [CWR] My name is Wesley Rowley. I am currently a Fellow at the ASME. This affidavit is prepared to support Enclosure 2 to the FSD Appeal. A statement of my professional qualifications is attached to this affidavit as Attachment B.

7. [CWR] Since 1989, I have been a member of the ASME Board on Nuclear Codes & Standards, which is jointly responsible for guiding the 1000 technical volunteers on eight standards committees, including the Committee on Operations & Maintenance of Nuclear Power Plants. I have been a member of the Operations & Maintenance Main Committee since 1984 and a member of the Subgroup ISTE since 2000. The O&M Committee develops the ASME OM Code and the ASME OM Standards/Guides.

8. [CWR] With respect to this joint affidavit, I have included my initials within brackets (*i.e.*, [CWR]) immediately preceding those sections I prepared.

Mark A. Gowin

9. [MAG] My name is Mark Gowin. I am currently employed as the Corporate Codes Program Manager for Tennessee Valley Authority. This affidavit is

prepared to support Enclosure 2 to the FSD Appeal. A statement of my professional qualifications is attached to this affidavit as Attachment C.

10. [MAG] I have thirty-one years experience in design, construction, modification, and testing of BWR and PWR systems. The last twenty-five years have been primarily focused on development and implementation of engineering programs to meet the requirements of ASME OM Code, ASME Section XI, and 10 CFR Part 50, Appendix J. In my current position at TVA I provide governance and oversight of ASME OM Code Inservice Testing (IST) Programs for all TVA nuclear units. I am responsible for ensuring that the IST Programs meet Code and regulatory requirements and industry best practices.

11. [MAG] With respect to this joint affidavit, I have included my initials within brackets (*i.e.*, [MAG]) immediately preceding those sections I prepared.

Statement of Expert Opinion

12. [All] The purpose of this affidavit is to demonstrate that the technical and factual information within my area of expertise and relied upon by TVA in the FSD Appeal is consistent with my professional judgment and expert conclusions.

13. [All] I have reviewed the Letter from NRC to TVA, "Final Significance Determination of a Red Finding, Notice of Violation, and Assessment Follow-up Letter (NRC Inspection Report 05000259/2011008) Browns Ferry Nuclear Plant," dated May 9, 2011. I have also reviewed the American Society of Mechanical Engineers (ASME), Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 1995 Edition, 1996 Addenda, Sections ISTC 4.1 and ISTC-

4.2.3. I am also familiar with the IST program at Browns Ferry Nuclear Plant, Unit 1.

14. [SMU, CWR] Based on my review of the BFN IST program and my professional judgment, I conclude that the BFN IST Program is in compliance with the American Society of Mechanical Engineers (ASME) Operation and Maintenance of Nuclear Power Plants Code (OM Code), 1995 Edition, 1996 Addenda, Paragraphs ISTC-4.1 and ISTC-4.2.3. The BFN IST methods are also supported by ASME Code Interpretation 99-9, which clarifies that supplemental indications are not a requirement of the ASME OM Code to verify valve obturator position.

15. [SMU] Based on my review of the BFN IST program and my expertise as author of NUREG-1482, Revision 1, I further conclude that the BFN IST program is consistent with NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Revision 1.

16. [MAG] I have determined through benchmarking with other plants that BFN's IST Program approach for implementing ISTC-4.1 and ISTC-4.2.3 is consistent with industry practices, including the IST program at its "sister" plant.

17. [SMU] I actively participated in the drafting of the FSD Appeal for TVA. I provided technical input related to IST program adequacy, including compliance with the applicable ASME OM Code requirements for IST Programs and NUREG-1482, Revision 1. I have reviewed those sections of the appeal addressing IST program adequacy and compliance with ASME Code requirements and NUREG-1482 and fully support the discussion provided therein.

18. [CWR] I actively participated in the drafting of the FSD Appeal for TVA. I provided technical input related to IST program adequacy, including compliance with the applicable ASME OM Code requirements for IST Programs. I have reviewed those sections of the appeal addressing IST program adequacy and compliance with ASME Code requirements and fully support the discussion provided therein.

19. [MAG] I actively participated in the drafting of the FSD Appeal for TVA. I provided technical input related to IST program adequacy, including benchmarking of the BFN IST program. I have reviewed those sections of the appeal addressing IST program adequacy and benchmarking and fully support the discussion provided therein.

20. [All] I hereby certify under penalty of perjury that the foregoing is true and complete to the best of my knowledge, information, and belief. Executed on June 8, 2011.

Executed in accord with 10 C.F.R. § 2.304(d),

/s/ Steven Unikewicz

Steven Unikewicz
Principal Engineer
Alion Science & Technology
1577 Spring Hill Rd, Suite 450
Vienna, Va 22182
(703) 439-7133
sunikewicz@alionscience.com

Executed in accord with 10 C.F.R. § 2.304(d),

/s/ C. Wesley Rowley

C. Wesley Rowley, PE
Vice President
The Wesley Corporation
P. O. Box 747
Green Valley, AZ 85622
(520) 777-8941
cwrowley@aol.com

Executed in accord with 10 C.F.R. § 2.304(d),

/s/ Mark A. Gowin

Mark A. Gowin
Corporate Codes Program Manager
Tennessee Valley Authority
3159 Vista Dr.
Cleveland, TN 37312
(423) 503-5931
mgowin@isi-ist.com

STEVEN M UNIKEWICZ
43177 Glenelder Terrace
Ashburn, VA 20147
(703) 439-7133
E: unikewiczs@asme.org

OBJECTIVE: To apply my formal education and practical experience in an Engineering, Licensing or Engineering Management capacity to enhance the production and use of water and energy. Particular interest in new nuclear plant development design, construction and operation.

SUMMARY: Work experience shows extensive experience in the field of Nuclear Power and Process Plant Regulation, Engineering, Design, Construction, Operation and project management. Performed multi-million dollar, multi-discipline projects on time and within budget. Performed complex regulatory and design issues in a complete, efficient manner. History of increasing leadership roles with employers, the technical community and the local community.

AWARDS:

USNRC Special Achievement Award, Palo Verde Inspection, April 3, 2006
USNRC Performance Awards, December 2003, 2004 and 2005
USNRC Special Act Award, Support of NRC Regional Offices, August 24, 2005
USNRC Special Act Award, GSI 191 Safety Evaluation Report, March 29, 2005
USNRC Region III Special Act Award, Kewaunee Special Inspection, August 4, 2004
USNRC Special Act Award, TMI Special Inspection, September 16, 2003
USNRC Special Act Award, September 9, 2003
American Society of Mechanical Engineers (ASME) 2002 Dedicated Service Award
Northeast Utilities/ASME Distinguished Engineer of the Year 2000

WORK EXPERIENCE

January 2008 To Present — Alion Science and Technology, Vienna, Va
Principal Engineer, Program Manager

Responsible for component and plant system evaluations and analysis. Technical areas include NPSH, gas entrainment in fluid systems, pump minimum flow protection, Plant Operability assessments, in-service testing of pumps and valves and Regulatory Review and support on all aspects of plant design and operation. Responsible for providing expert consulting services in the areas of GSI-191 Ex and In-Downstream Effects and Chemical Effects to the nuclear power industry.

Specifically, wrote the MHI(Mitsubishi Heavy Industries) Downstream Effects (ex-vessel) report for the US-APWR new reactor submittal to the USNRC including their white paper on NPSH. Provided training and consultation to the Korean Electric Power Research Institute on the development and Implementation of their AOV and Fire Protection programs. Provided research and technical support to St. Lucie Nuclear station on their response to NRC GL 08-01 and on Instrument Air issues. Provide NRC and NEI interface for MHI and other US Nuclear Utilities

Since June 2008 working with (badged) Calvert Cliffs Nuclear Station (CCNPP) and Nine Mile Point (NMP) containment walkdowns, Chemical Effects Issues and Downstream Effects evaluations. Supported CCNPP during the Spring 2009, 2010 and 2011 refueling outages. Trained in various Constellation programs and have written design changes and engineering packages.

Have consulted with numerous other nuclear plants and utilities on air entrainment, system flow balancing and other plant design and operational issues.

Act as Alion interface with NRC as needed.

July 2002 US Nuclear Regulatory Commission, Washington DC
To
January 2008 Senior Mechanical Engineer, Office of Nuclear Reactor Regulation, Mechanical Branch

Member, USNRC Headquarters Incident Response Team, Reactor Safety Team counterpart to Protective Measure Team

Lead reviewer of USNRC GSI-191, Containment Sumps" downstream effects". This includes systems, components and reactor vessel reviews. Mechanical Engineer involved in the review and development of Codes, Standards and Regulations regarding the In-Service Testing of Components at Nuclear Power Generating Stations. Author of NUREG-1482, rev 1 Guidelines for Inservice Testing at Nuclear Power Plants. Principal reviewer and writer of the Davis Besse High Pressure Injection System and Pump modifications USNRC Safety Evaluation. NRC lead on Emergency Diesel Generator, air operated valve and pump issues. Branch lead on pump, valve and EDG design, operation, and testing issues. Provide technical leadership and guidance to NRC Headquarters Staff, and Regional and Site Inspectors and to the power industry. Member of two ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) Code Committees; OM16 Diesel Drive Working Group and OM19 Air Operated Valves. NRC/NRR contact and technical representative to OM ISTB, Committee on Pumps. NRC representative to three industry users groups; Air Operated Valve Users Group (AUG), Emergency Diesel Generator Users Group, and the Inservice Test Owners Group (ISTOG) and frequently interact with the Nuclear Energy Institute (NEI) on various technical and policy issues. Routinely write technical and safety evaluations for ASME Code relief requests and proposed license amendments to site Technical Specifications. Lead NRR Division of Engineering mechanical reviewer for issues pertaining to Emergency Diesel Generators and heat exchangers (non-steam generator). Writer of numerous USNRC generic communications. Technical reviewer for Westinghouse AP1000 DCD. Involved with review of the other new reactor designs submitted to USNRC.

NRR/DCI representative to the Operating Experience (OpE) program. Routinely involved in special inspections of critical systems at US commercial sites. Developed and presented formal classroom training, as an instructor to NRC Region Offices and Inspectors on Inservice Testing and on the design, operation and testing of air operated valves. Act as Division Branch Chief as needed.

Nov 2007 Fru-con Construction, Woodbridge, Va.
to
May 2009 Part-time Start-up Engineer

Responsible for preparing and implementing plan to start-up and commission the Broad Run Waste Reclamation Facility and Kent Narrows Waste Facility and supporting start up and testing at the Arlington, Va. and New Design Road, Fredrick Md. facilities.

February 2002 Modern Continental South, Griffith Water Treatment Plant, Lorton Va.
To
October 2006 Chief Engineer/Start-up Engineer

Lead Engineer for the construction of the 128 MGD, $120m water treatment facility. Supervised a 6 person multi-discipline engineering and design staff. Supported field construction, resolved design conflicts, specified and purchased mechanical and chemical systems equipment.

Mechanical Field / Start-up Engineer for the construction of the 128 MGD, $120m water treatment facility. Author and principal developer of the start-up plan. Wrote test plans, test procedures and supervised the testing of mechanical and fluid systems. Supported field construction and resolved design conflicts.

Part-time Start-up Engineer July 2002 to commissioning in Oct 2006.

Author of the Start-up Plan and operations guidelines for the 11 Chemical Feed Systems, Flocculation / Sedimentation and Filtration System and the Finished Water Pumping System. Supervised final submittal preparation, O&M Manual preparation and final closeout and beneficial use documentation. Acted as pump installation, test and commissioning specialist.

April 2001 To January 2002 — Energy Education Inc, Wichita Falls, TX
Consulting Engineer

Provided technical advice and guidance on the use and conservation of energy and utilities resources including proper specification, design and operation of boiler and HVAC systems for School Districts. Trained clients in the interpersonal and technical aspects of the EEI program. Guided clients in the management of people and the operation of complex systems.

May 1992 To April 2001 — Northeast Utilities, Waterford, CT
Position: Senior Engineer – Millstone Unit 2 Design Engineering

Accident Management Team – Mechanical Engineer, Station Emergency Response Organization. Member, Nuclear Safety Assessment Board, Engineering Sub-Committee.

6/99 to 4/01 — Technical and Project lead for mechanical and fluid system design and technical issues at Millstone Unit 2. Specifically, prepared and implemented design changes, including cost benefit analysis, budget management, schedule development, and coordination of all project aspects relating to projects and design changes. Lead Engineer for station evaluations and modifications.

Projects included Millstone 3, Turbine Long Shank Buckets, ASME VIII Vessel erosion/corrosion repairs in Millstone 2 Secondary Systems. Peer mentor for junior engineering and design personnel. Discipline Engineer for mechanical systems and component modifications. Wrote Safety Evaluations, Technical Specification Changes, FSAR Changes and detailed reviews and analysis of systems and components.

Chair of the Millstone Station Design Control Manual (DCM) Working Group. The DCM guides the engineering and design processes at the station.

4/98 to 6/99 — Acting Supervisor – MP2 MEPL Program Group (Material, Equipment & Parts – Q list). Responsible for the maintenance, technical evaluation, and implementation of the MEPL Program and Q-list at Millstone 2.

9/97 to 4/98 — Acting Supervisor, Plant Engineering Group. The group was comprised of 25 Engineers and designers responsible for the field implementation of all plant modifications to Millstone Unit 2. The group was multi-discipline (mechanical, civil, I&C, and elect). We implemented in excess of 80 large scale modifications in 8 months.

5/97 to 9/97 — Projects Lead for Task Managed Mechanical Design and Modifications for Millstone 2. Responsible for the design of over 30 large scale design modifications with an engineering budget of $6.5 million. Projects managed were performed by Raytheon Engineers, Stone and Webster, Proto-Power Corporation, and Duke Engineering.

11/96 to 5/97 Acting Supervisor, System Design Engineering. Responsible for the design, evaluation, and modification of all mechanical plant systems including the responsibility of the 10CFR50 Appendix R Program.

5/92 to 11/96 Senior Engineer, Millstone Unit 2 Design Engineering. Technical lead for all RBCCW issues, GL 96-06 concerns and all other issues associated with Millstone 2. Senior Project and Mechanical Systems Engineer for projects at Millstone 2. Specifically prepared and implemented design changes, cost benefit analysis, budget and schedule. Managed and coordinated all aspect relating to projects, their evaluation and their implementation. Large projects included the redesign of the containment sump Recirculation System; Unit Station and Instrument Air Upgrades; Spent Fuel Storage Capacity Increase; Demonstration of NFBC Compaction; a new Nitrogen Storage and Feed Facility for the Condensate Storage Tank and House Loads; CRAC System Modifications, redesign of the RCP Motor Lube Oil Collection System.

Lead and Mechanical Engineer for mechanical systems and component and station evaluations and modifications. Performed detailed reviews and analysis of systems and components. Performed Safety Evaluations and made changes to Technical Specifications, FSAR, etc.

Aug 1989 To May 1992 Northeast Utilities, Berlin CT

Position: Engineer, Fossil Plant Programs, Fossil Hydro Engineering and Construction

Life Extension/ Non Destructive Evaluation coordinator for NEU's nine (9) reheat power plants. Developed and implemented NDE Programs to evaluate and monitor all high energy, thick-walled, boiler and piping components.

Developed and implemented NDE programs to assess the condition of power plant heat exchangers. Mechanical discipline engineer for mechanical systems and components analysis, evaluation and modification.

July 1985 to Aug 1989 Northeast Utilities, Berlin CT

Position: Engineer, Balance of Plant Systems, Generation Mechanical Engineering

Responsible for the engineering, design, analysis, procurement and installation of modifications to NEU Nuclear Power Stations (Connecticut Yankee and Millstone Units I, II, and III). Major Projects included the analysis of the CY Service Water System to ensure compliance with design and operational requirements; the redesign, procurement and installation of the Millstone II Condenser (900 MWe unit).

June 1982 To July 1985 Connecticut Yankee Atomic Power Co, Haddam Neck CT

Position: Engineering Technician

Assistant to Engineering Manager providing research and support of special projects. Developed the Stops/Seals installation and inspection program. Performed evaluations and design changes to plant systems and components. Performed performance monitoring of reactor fuel burn-up and reactor core performance. Certified ISI Level II Examiner. Performed vibration monitoring of critical plant equipment.

PUBLICATIONS (sample)

ASME Energy Talking Point (ETP) 1, Three Signs the End of Oil Exports is Coming, Feb 03, 2011

ASME Position Paper PS11-04, Expanding Nuclear Power in the United States, January 2011

Technical Reviewer, Nuclear Engineering Handbook, K.D. Kok CRC Press 2009

NUREG 1482, Guidelines for Inservice Testing at Nuclear Power Plants, Rev. 1, January 2005

Speaker/Moderator ASME Climate Change Workshop, Washington DC, July 15, 1999

Fluid Transients in a Voided System, co-author, ASME Fluids Division, Nov 1998

Implementation of Spent Fuel Storage options – A Utility Project Engineer's Perspective, ANS/ASCE High Level Radioactive Waste Management Conference LV, NV 5/22/94

Guest Lecturer, George Washington University, School of Graduate Studies, Nov 2009, Nov 2010

Speaker/Moderator NRC/ASME 9th Pump and Valve Symposium, Washington DC, July 18, 2006

Speaker – NRC/NEI Joint Workshop – NRC Generic Letter 96-06 Thermal Overpressurization of Piping, Gaithersburg, MD, Dec 4, 1997

Speaker, Institute of Nuclear Materials Management, Spent Fuel Pool Storage Options, Washington DC, Jan 27, 1994

Panelist – NRC Generic Letter 96-06 Thermal Overpressurization of Piping, ASME International Joint Power Conference, Nov 4, 1997

PROFESSIONAL SOCIETIES
Member, American Society of Mechanical Engineers
Chair, ASME Energy Committee
Member, ASME OM19 Code Committee, Air Operated Valves
Member, ASME OM Code, New Reactors Sub-Group
Member, ASME Plant Operations and Design Committee
Former Member, ASME OM16 Code Committee, Diesel Drive Working Group
Former Member, Board on Pre-College Education, Council on Education
Former Chair, Systems & Power Generation Facilities Design Committee, Power Division

EDUCATION: University of Hartford, West Hartford CT
Bachelor of Science – Mechanical Engineering

TRAINING
USNRC Incident Investigation Team, qualified 2/14/03
ASME Leadership Development Intern, Council on Engineering, 1995 - 1996
ASME Short Course – Interpretation and Evaluation of Weld Defects, March 1990
Formerly Certified ASME Level II Visual Examiner VT I, II, III, IV
Formerly Certified Asbestos Supervisor, Certification # 421 P & P/S

PUBLIC SERVICE
Colchester CT, Board of Education Member, Secretary 1998 – 2001
Colchester CT, Water and Sewer Commission, Member 1992 - 1998

REFERENCES Available upon request

ASME Professional Resume of C. Wesley Rowley, PE

Education:

BS in General Engineering, University of Illinois, 1965
MS in Nuclear Engineering, University of Illinois, 1967
MA in Strategic Studies, Naval War College, 1986

Current Experience:

1976-pres ASME International, Fellow (since 1998)

1989-pres Member, Board on Nuclear Codes & Standards
Jointly responsible for guiding the 1000 technical volunteers on six standards committees and three Boiler & Pressure Vessel subcommittees:

- Committee on Nuclear Air & Gas Treatment
- Committee on Operations & Maintenance of NPPs
- Committee on Cranes for Nuclear Facilities
- Committee on Nuclear Quality Assurance
- Committee on Qualification of Mechanical Equipment
- Committee on Nuclear Risk Management
- B&PV Subcommittee on Nuclear Design
- B&PV Subcommittee on Inservice Inspection
- B&PV Subcommittee on Nuclear Accrediation

1976-pres Member of Operations & Maintenance Main Committee (since 1984); member of the Subgroup ISTE (since 2000). The O&M Committee develops the ASME OM Code and the ASME OM Standards / Guides.

2001-pres Member of the Post Construction Main Committee (since 2002); also member of the PCC Subcommittee on Repair & Testing (since 2001); member of the Non-metallic Repair Project Team (since 2001); chairman of the Non-metallic Repair Project Team (since 2002). The PCC develops PCC-1, PCC-2, and PCC-3 standards.

2003-pres Chairman of BPV / Subcommittee II, Materials / Special Working Group, Non-Metallic Materials. The SWG-NMM is responsible for BPV Section II / Part E, Non-metallic Materials.

2006-pres Member of BPV / Subcommittee III, Nuclear Power / Special Working Group – Polyethylene Pipe. The SWG-PP is responsible for Code Case N755.

2006-pres	Member of Nominating Committee (chairman since 2007)
2006-pres	Co-chairman of the Nuclear Risk Management Coordinating Committee. Responsible for coordinating risk management issues between the technical societies (e.g., ASME, ANS, and IEEE).

Past Experience:

1989-pres	Past chairman of BNC&S (2002-2005); past member of Committee on Board Operations (2000-2002); past member of Committee on Honors and Awards (1997-2002); past chairman of Board Risk Mgmt Task Group 1999-2002); past chairman of Board Strategic Planning for Facilities 1990-1993.
1976-pres	Previous member of the O&M Executive Committee (1987-2002); previous chairman and member of O&M Special Committee on Standards Planning (1990-2002); previous chairman and member of the Electrical Subgroup (1982-1984); previous member of the O&M Subcommittee on Performance Testing (1976-89); previous member of the O&M Closed Cooling Water Systems Project Team (published OM-2-1982).
2003-2006	Past chairman of Joint BPV III-XI Project Team on Plastic Pipe
2002-2005	Past member, Council on Codes & Standards; past member of CC&S-TF on Performance-Based Standards

Mark A. Gowin
3159 Vista Dr.
Cleveland, TN 37312
Phone: 423-503-5931 Email: mgowin@isi-ist.com

SUMMARY OF WORK EXPERIENCE

Thirty one years experience in design, construction, modification, and testing of BWR and PWR systems. The last twenty five years have been primarily focused on development and implementation of engineering programs to meet the requirements of ASME OM Code, ASME Section XI, and 10CFR50, Appendix J.

ASME OM Code Inservice Testing Programs

- Provide governance, oversight, and technical support of ASME OM Code Inservice Testing (IST) Programs for all TVA nuclear units to ensure the IST Programs meet Code, regulatory, and industry best practices.
- Develop and implement ten-year updates of numerous Inservice Testing Programs (Pumps and Valves) to incorporate the latest edition and addenda of OM Code endorsed by NRC. This included preparation of program documents, relief requests, and implementation procedures.
- Conduct multi-day training classes for Inservice Testing Program Owners including development of training materials
- Perform pump and valve tests
- Provide technical support for resolution of Inservice Testing Program and component related issues.
- Audit/Assessment of Inservice Testing Programs
- Regular attendance of OM Code Committee meetings

Appendix J Containment Leak Rate Testing Programs

- Provide governance, oversight, and technical support of 10CFR50.55a, Appendix J, Containment Leak Rate Testing Programs for all TVA nuclear units to ensure the IST Programs meet Code, regulatory, and industry best practices.
- Develop and implement Appendix J Leak Rate Programs including program documents and implementation procedures.
- Perform Type "B" and "C" containment local leak rate tests
- Perform Type "A" containment integrated leak rate tests

ASME Section XI Inservice Inspection Programs

- Develop and implement ASME Section XI IWE Containment Inservice Inspection Programs including program documents, relief requests, and drawings
- Develop and implement ASME Section XI Pressure Test Programs including program document, relief requests, and drawings
- Development of a Repair/Replacement Work Order/NIS-2 tracking database for Browns Ferry Unit 1 Recovery.
- Assist in development and implementation of ASME Section XI Inservice Inspection Programs including drawings, component counts, and database update.

Check Valve and Relief Valve Testing Programs

- Develop and implement plant wide check valve and relief valve programs with scope which include and extend beyond the ASME Inservice Testing Program requirements.

Engineering and Design

- Project engineering for plant modifications including preparation of design changes, 10CFR50.59 review, development of implementation procedure work order, resolution of field installation problems, and post-modification testing.

SUMMARY OF WORK EXPERIENCE (Continued)

Engineering and Consulting Business Owner (Prior to employment at TVA)

Founder and owner of Program Engineering and Consulting, Inc. (PEC, Inc.) which specializes in development, implementation, and audit/assessment of engineering programs such as, Inservice Inspection, Inservice Testing, and Appendix J. PEC, Inc. has been in business since February 2002 and provided services to Bruce Power (Canada), Eskom (South Africa), Florida Power and Light, Nuclear Management Company, Progress Energy, and Tennessee Valley Authority. PEC, Inc. has engaged in short term and long term fixed-price contracts and staff augmentation contracts with peak annual revenue of one million dollars. Responsibilities include: marketing and business development; preparation of bid proposals; execution and maintenance of contracts; customer relations; invoicing/billing; payroll; compliance with State and Federal regulations and taxes (unemployment, excise, income, FICA, Medicare, etc.); Project management (budget, schedule, manpower) of simultaneous projects involving up to 6 full-time and 3 part-time personnel.

INDUSTRY PARTICIPATION

- Attend and participate in ASME OM Code meetings since 2002. Current member of Subgroup ISTB
- Member of Inservice Testing Owner's Group. Attend meetings and participate in working group activities.
- Attend and participate in NRC/ASME Symposiums on Pump and Valve Testing
- Attend ANSI/ANS 56.8, Containment System Leakage Testing requirements, committee meetings and participate in working group activities
- Member of Appendix J Owner's Group. Host of 2010 annual meeting and participate in group activities

EDUCATION / CERTIFICATION

- Cleveland State Community College - 1976 - 1977
- University of Tennessee Knoxville - 1977 - 1978 (completed engineering courses such as calculus, statics, dynamics, etc.; degree not completed)
- ANSI 45.2.6 Level II Appendix J Testing
- Various utility sponsored training courses such as: BWR Systems; Root Cause Analysis; 10CFR50.59 Evaluation; Independent Qualified Review; Design Change/Modifications
- Experience level and training required to obtain TVA Qualification Cards for Inservice Testing Program, Appendix J Program, Pressure Test Program, and Containment Inservice Inspection Program.

DETAILED WORK EXPERIENCE

1/08 to Present — *Corporate Codes Program Manager* Tennessee Valley Authority
Corporate Office - Chattanooga, TN

- Provide governance and oversight of ASME OM Code Inservice Testing (IST) Programs for all TVA nuclear units. Responsibility to ensure the IST Programs meet Code, regulatory, and industry best practices.
- Provide governance and oversight of 10CFR50, Appendix J, Containment Leak Rate Test Programs for all TVA nuclear units. Responsibility to ensure the Appendix J Programs meet Code, regulatory, and industry best practices.
- Provide technical authority for to all TVA nuclear units in matters related to IST and Appendix J.

3/02 to 1/08 *Sr Program Engineer*, Program Engineering and Consulting, Inc.

Browns Ferry Nuclear Plant

- Appendix J Program Engineer responsible for development and implementation of Unit 1 Appendix J Containment Leak Rate Testing Program. This includes preparation of the Appendix J program plan, review of associated design change notices and work orders, review of implementing test procedure revisions and review of completed test data.
- IWE Program Engineer responsible for development and implementation of Unit 1 IWE program. This includes preparation of the IWE program plan, relief requests, scan plan, review of associated design change notices and work orders, coordination/scheduling of inspection activities and resolution of program related issues.

Shearon Harris Nuclear Plant

- Prepared ten year update of Inservice Testing (IST) Program to 2001 Edition/2003 Addenda of ASME OM Code.
- Determined IST Program scope and test requirements.
- Developed relief requests, cold shutdown justifications and refueling outage justifications.
- Prepared implementing test procedure mark-ups.

Koeberg Nuclear Power Station

- Prepared ten year update of Inservice Testing (IST) Program to 2001 Edition/2003 Addenda of ASME OM Code.
- Determined IST Program scope and test requirements.
- Developed relief requests, cold shutdown justifications and refueling outage justifications.

Bruce Nuclear Development Plant

- Developed Check Valve Condition Monitoring Program for Units 1 – 4 similar to Units 5 – 8 below.

7/01 to 3/02 *IST Engineer*, BCP Technical Services

H. B. Robinson Nuclear Plant

- Prepared ten year update of Inservice Testing (IST) Program to 1995 Edition/1996 Addenda of ASME OM Code.
- Determined IST Program scope and test requirements.
- Developed relief requests, cold shutdown justifications and refueling outage justifications.
- Wrote/revised all associated IST Program administrative procedures including safety/relief valve testing program and check valve condition monitoring program.
- Reviewed all surveillance test procedure revisions to ensure compliance with OM Code test requirements and ensure all IST Program required tests captured.

2/00 to 7/01 *IST Technician*, Onsite Engineering Services

Bruce Nuclear Development Plant

- Develop a Check Valve Condition Monitoring Program for Units 5 - 8.
- Prepare Check Valve Program procedures.
- Participate in Expert Panel to recommend preventive maintenance activities for check valves.
- Develop Microsoft Access database for the Check Valve Program and Basis Document.

7/99 to 2/00 *IST Engineer*, BCP Technical Services
D.C Cook Nuclear Plant
- Performed duties as Safety and Relief Valve Testing Program Manager such as:
- Developed a Safety and Relief Valve Testing Program.
- Coordinated offsite testing of Main Steam Safety Valves and Pressurizer Safety Valves.
- Scheduled and coordinate onsite testing of safety and relief valves.

St. Lucie Nuclear Plant
- Resolved NRC request for additional information for the IST Program update.
- Prepared IST Program revision (including new relief request and refueling outage justifications) to incorporate plant modifications and NRC request for additional information.

8/98 to 7/99 *IST / ISI Engineer*, Enertech Servus
Various Nuclear Plants
- Assisted in development of Risk Informed Inservice Inspection Program based on the methodology described in ASME Section XI, Code Case N-577 and WCAP-14572.
- Developed comprehensive pump tests in accordance with ASME Section XI, 1995 Edition.
- Performed independent ISI /IST assessment.

10/97 to 8/98 *IST Engineer*, Enertech Servus
Shearon Harris Nuclear Plant
- Performed duties as IST Coordinator on temporary basis until permanent position filled.
  - Reviewed all IST related test performance packages for acceptability and test data trending.
  - Reviewed design change packages and procedure revisions for IST compliance.
  - Resolved day-to-day IST related technical issues.
  - Reviewed work orders and assigned post-maintenance testing requirements.
- Prepared ten year update of the Inservice Testing Program.
  - Determined IST Program scope and test requirements.
  - Developed relief requests, cold shutdown justifications and refueling outage justifications.
  - Prepared IST Program Plan, administrative procedures and 10CFR50.59 safety evaluations.
  - Developed IST Basis Document to justify component inclusion/exclusion in IST Program.
  - Developed Microsoft Access database for the IST Program and IST Basis Document.

6/97 to 10/97 *IST / ISI Engineer*, Enertech Servus
Unit 3 Millstone Nuclear Plant
- Developed ASME Section XI Pressure Test Program for the second ten-year interval.
  - Developed individual pressure test boundaries that optimize the use of normal plant operating conditions and surveillance tests.
  - Developed Microsoft Access database to correlate individual test boundaries with specific code requirements and relief requests (e.g., test pressure, hold time, VT-2 boundary).
  - Wrote generic surveillance procedure to perform all periodic pressure tests.

1/96 to 6/97 *IST Program Coordinator*, System Engineering Department
Vermont Yankee Nuclear Plant

- Performed activities associated with the ASME Section XI Inservice Testing Program including development and maintenance of program document, oversight of program implementation and maintaining awareness of industry/regulatory issues.
- Provided day-to-day assistance and technical input to Operations, Design Engineering and Maintenance personnel related to IST activities.
  - Reviewed surveillance procedures, design changes and LCO maintenance plans.
  - Prepared safety evaluations and Basis for Maintaining Operability determinations.
- Prepared Event Reports including investigation, root cause analysis and corrective action.

Sr. Engineer, System Engineering Department

- Developed ASME Section XI Inservice Testing Program Basis Document.
- Performed major revision to IST Program to incorporate changes identified during development of the basis document. This revision included program scope changes, test type and frequency changes, new/revised cold shutdown and refueling outage justifications and new relief requests.
- Designed Microsoft Access database for the IST Program and Basis Document.

4/95 to 12/95 *Sr. Engineer,* Contracted to Tennessee Valley Authority, Technical Support Section, NSSS Systems
Sequoyah Nuclear Plant

- Wrote/revised surveillance testing procedures for ten year update of ASME Section XI Inservice Testing Program (1989 Edition, OMa-1988, Parts 1, 6 and 10).

8/94 to 4/95 *Sr. Engineer,* Contracted to Mechanical Engineering and Construction Department
Vermont Yankee Nuclear Plant

- Project Engineer for containment modification project. These duties included:
  - Prepared design change package, installation and testing procedure, 10CFR50.59 evaluations, labor/material cost estimates, project budget and project schedule.
  - Coordinated modification implementation with other plant activities.
  - Ensured modification completion within budget and schedule requirements.
- Trained as Root Cause Evaluator.

1/93 to 4/94 *Sr. Mechanical Technologist,* Contracted to Tennessee Valley Authority
Watts Bar Nuclear Plant

- Wrote/revised surveillance testing procedures for ten year update of ASME Section XI Inservice Testing Program (1989 Edition, OMa-1988, Parts 1, 6 and 10).

9/92 to 1/93 *Mechanical Engineer,* Contracted to Carolina Power and Light, Nuclear Engineering
Brunswick Nuclear Plant

- Prepared installation and testing portions of design change packages.
- Prepared procedures for installation and acceptance testing of modifications.
- Certified as Design Verifier and Qualified Safety Reviewer (10CFR50.59).

2/92 to 9/92 *Mechanical Engineer,* Contracted to Gulf States Utilities, NSSS Design Engineering
River Bend Station

- Prepared ASME Section XI check valve disassembly and inspection procedures.
- Prepared design change packages and 10CFR 50.59 safety evaluations.
- Performed field engineering for implementation of modifications.
- Closure of design change packages and various other outstanding items.

8/86 to 2/92 *Mechanical Engineer,* Contracted to Tennessee Valley Authority, Systems Engineering
Sequoyah Nuclear Plant

- Prepared procedures to satisfy Technical Specification surveillance requirements such as:
  - ASME Section XI pump tests, valve tests, and inservice pressure tests.
  - Emergency Core Cooling System (ECCS) performance/flow balance tests.
  - 10 CFR 50, Appendix J containment leak rate tests.

5/84 to 8/86 *System Engineer,* Bechtel Corp, Georgia Power Company
Plant Vogtle

- Conducted NSSS and BOP systems walk downs, initiated work authorizations and expedited design and field activities to ensure systems completion and start-up per project schedules.
- Performed deviation report (non conformance item) evaluation and closure.

12/81 to 5/84 *Field Engineer,* Bechtel Corp., Mechanical Engineering Group
Enrico Fermi II

- Developed and implemented ASME Section XI repair/replacement and testing packages.
- Performed troubleshooting and maintenance engineering for pumps and valves.
- Prepared design change packages for modifications and upgrades.
- Conducted functional tests and hydrostatic/pneumatic pressure tests.

2/81 to 12/81 *Field Engineer,* Mercury Co., Louisiana Power and Light
Waterford III

- Designed ANSI B31.1, ASME Class 2 and ASME Class 3 small bore piping, instrument tubing and instrument installations for NSSS and BOP systems.

1/80 to 2/81 *Designer,* Contracted to Tennessee Valley Authority, Division of Engineering and Design
Bellefonte Nuclear Plant

- Generated instrumentation loop schematics, flow diagrams, installation details, control panel layouts and other I & C related drawings.

ENCLOSURE 3

Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Reply to a Notice of Violation, EA-11-018

ENCLOSURE 3

Browns Ferry Nuclear Plant, Unit 1
Docket No. 50-259

Reply to a Notice of Violation, EA-11-018

Restatement of Violation

During an NRC inspection completed on December 31, 2010, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Browns Ferry Nuclear Plant Unit 1 Technical Specification (TS) LCO 3.5.1, Emergency Core Cooling System (ECCS) — Operating, requires, in part, that each ECCS injection/spray subsystem shall be operable in Modes 1, 2, and 3. Action statement Condition A states that with one low pressure ECCS injection/spray subsystem inoperable, restore the low pressure ECCS injection/spray subsystem to operable status within seven days. Action statement Condition B states that with the required action and associated completion time of Condition A not met, be in Mode 3 within 12 hours and in Mode 4 within 36 hours.
>
> Contrary to the above, from March 13, 2009 to October 23, 2010, a Unit 1 low pressure ECCS injection/spray subsystem was inoperable while in Modes 1, 2, and 3, and the licensee failed to restore the subsystem to operable status within seven days, or complete Action statement Condition A and B within the required time. Specifically, the Unit 1 Residual Heat Removal Loop II subsystem was inoperable, because the licensee failed to maintain the Unit outboard Low Pressure Coolant Injection (LPCI) valve 1-FCV-74-66 in an operable condition, which rendered a low pressure ECCS injection/spray subsystem (the RHR loop II subsystem) inoperable while Unit 1 was operating in Mode 1.

This violation is associated with a Red significance determination process finding for Unit 1 in the Mitigating Systems cornerstone.

Admission or Denial

TVA admits the Technical Specification operability violation.

Reasons for the Violation

TVA determined that the root cause of the valve failure was an original manufacturing deficiency. Separation of the valve disc from the disc skirt/stem assembly was due to

undersized skirt threads. Backpressure on the disc exceeded the strength of the threaded connection when the valve was cycled open, causing axial thread pull-out.

Corrective Steps Taken and Results Achieved

- TVA repaired valve 1-FCV-74-66 by reworking the skirt threads to their original design configuration to preclude disc separation. The RHR loop II subsystem was restored to operability.
- TVA verified that the valve discs were intact in all like valves, but was unable to unscrew the discs from the skirt/stems to verify proper thread size. TVA added welded gussets to the disc to skirt/stem interface on all like valves to preclude disc separation.
- TVA implemented controls limiting back-pressure on the valves.

Date of Full Compliance

Valve 1-FCV-74-66 was repaired and restored to operable status during the Unit 1 refueling outage that ended November 23, 2010. Other like valves were examined and modified during shutdowns of all three units that ended as follows: Unit 1 on May 26, 2011; Unit 2 on April 2, 2011; and Unit 3 on May 31, 2011.