



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

April 8, 2011

EA-11-018

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: PUBLIC MEETING SUMMARY– REGULATORY CONFERENCE FOR
BROWNS FERRY NUCLEAR PLANT, DOCKET NO. 50-259

Dear Mr. Krich:

This refers to the meeting conducted on April 04, 2011, in Atlanta, GA. The purpose of this meeting, was to allow representatives of Tennessee Valley Authority (TVA), the licensee for Browns Ferry Nuclear Station, Unit 1, to meet with U. S. Nuclear Regulatory Commission (NRC) personnel to discuss one preliminary Greater Than Green finding associated with one Apparent Violation that was documented in NRC Inspection Report 05000259/2010005, 05000260/2010005, and 05000296/2010005 (ML110610641). The findings dealt with the failure to establish adequate design control and perform adequate maintenance on the Unit 1 low pressure coolant injection (LPCI) outboard injection valve, 1-FCV-74-66, resulting in the valve being left in a significantly degraded condition and residual heat removal (RHR) loop II unable to fulfill its safety function.

A list of attendees and a copy of your presentation handout are enclosed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Should you have any questions concerning this meeting, please contact me at (404) 997-4662.

Sincerely,

/Craig Kontz RA for/

Eugene F. Guthrie, Chief
Projects Branch 6
Division of Reactor Projects

Docket No.: 50-259
License No.: DRP-33

Enclosure: As stated

cc w/encl: (See page 2)

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Mr. Rod Krich
Vice President Nuclear Licensing
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X PUBLICLY AVAILABLE NON-PUBLICLY AVAILABLE SENSITIVE X NON-SENSITIVE
ADAMS: Yes ACCESSION NUMBER: ML111010106 SUNSI REVIEW COMPLETE

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NAME	CKontz	EGuthrie					
DATE	04/08/2011	04/08/2011					
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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2

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Letter to R. M. Krich from Eugene Guthrie dated April 8, 2011

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BROWNS FERRY NUCLEAR PLANT, DOCKET NO. 50-259

Distribution w/encl:

C. Evans, RII

L. Douglas, RII

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

REGULATORY CONFERENCE
 TENNESSEE VALLEY AUTHORITY
 APRIL 04, 2011
 NRC REGION II, ATLANTA, GEORGIA

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**TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT, UNIT 1**

Regulatory Conference

**Low Pressure Coolant Injection Valve
1-FCV-74-66**

Atlanta, Georgia

April 4, 2011

Enclosure 2



Agenda

- Introduction Preston Swafford
- Background Rob Whalen
- Root Cause/Engineering Analyses Rob Whalen
- Significance Determination James Emens
- Performance Deficiency James Emens
- Corrective Actions Rob Whalen
- Long-Term Fire Strategies Rob Whalen
- Closing Remarks Preston Swafford



Introduction

- On October 23, 2010, Residual Heat Removal (RHR) Loop II Injection Valve, 1-FCV-74-66, did not pass flow while operations was aligning shutdown cooling at the beginning of a refueling outage
- Disassembly of the valve revealed the disc separated from the stem and lodged in the seat
- The disc separation from the stem resulted from an original manufacturing defect, undersized threads in the disc skirt/disc assembly
 - Preliminary root cause was thought to be lack of skirt key caused disc separation, which was basis for apparent violation
 - Final root cause shows that the cause of the disc separation is not a licensee performance deficiency
- Based on results of extensive forensic examination, analysis, and laboratory mockup tests, we have shown conclusively that the disc would have released within an acceptable time with an RHR pump running (due to friction reduction from pressure pulsations), allowing the valve to provide functional flow
- TVA is taking steps to significantly reduce risk due to fire at the Browns Ferry Nuclear Plant
 - Reducing instances of Self-Induced Station Blackout (SISBO) actions
 - Accelerating plant changes identified as part of NFPA 805 transition
 - Changing the Safe Shutdown Instructions (SSIs) to allow the use of alternate shutdown paths



Background – Assessment Approach

- Root cause team was assembled including site and corporate expertise
- Comprehensive forensics were performed to determine root cause
 - Southwest Research Laboratory (weld examinations)
 - Westinghouse Laboratory (valve component forensics)
 - Structural Integrity (thread strength analysis, sensitivity study)
 - Independent Burns & Roe metallurgist (aggregate review of forensics reports)
- Performance Improvement International (PII) performed detailed analysis and laboratory testing to determine the valve's capability to function in its as-found state of separation

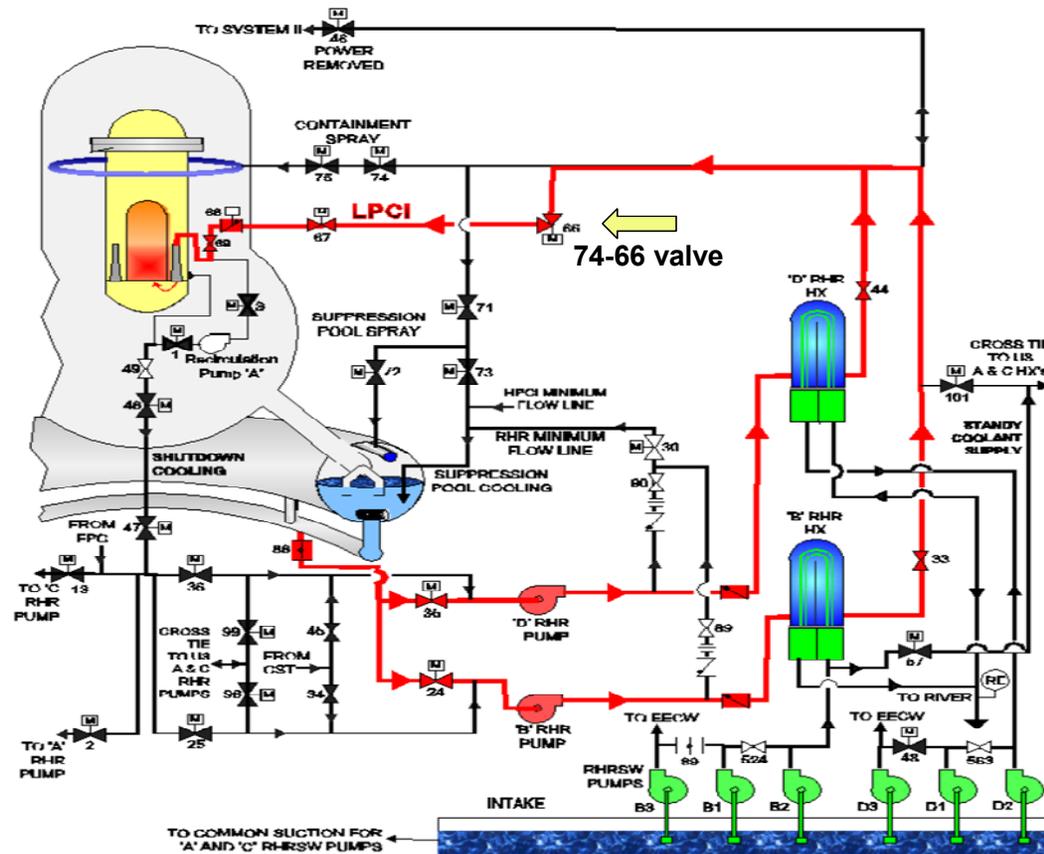


Background – Timeline

Timeline	
1968	Walworth valve purchased as an assembly from General Electric for construction of Browns Ferry Nuclear Plant (BFN), Unit 1
December 1974	Separation/loosening of discs due to flow-induced vibration
1983	Installed modified disc with “V” notch trim (skirt reused)
June 2006	Replaced stem prior to BFN, Unit 1, restart due to observed stem nut damage
2007 to October 2010	Satisfactory quarterly valve stroke times based on limit switch indication, not torque
March 2009	Initiated shutdown cooling passing 7,000 gpm flow through valve
October 2010	Initiated shutdown cooling with no observed flow (terminated pump operation after 110 seconds)

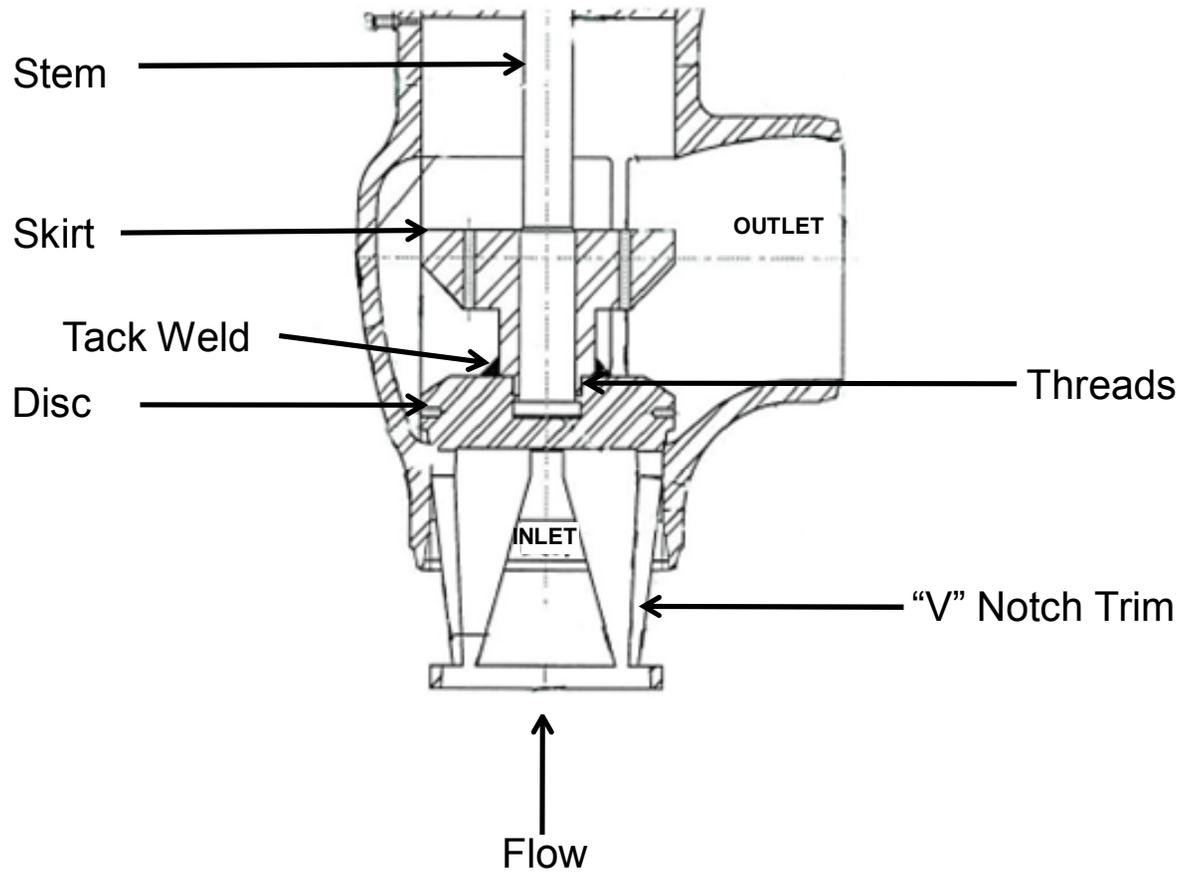


Background – RHR Low Pressure Coolant Injection (LPCI) System Overview





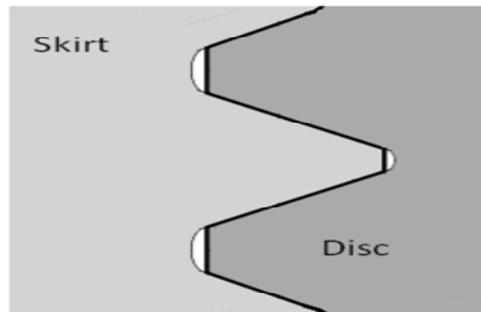
Root Cause – RHR Valve Cross Section



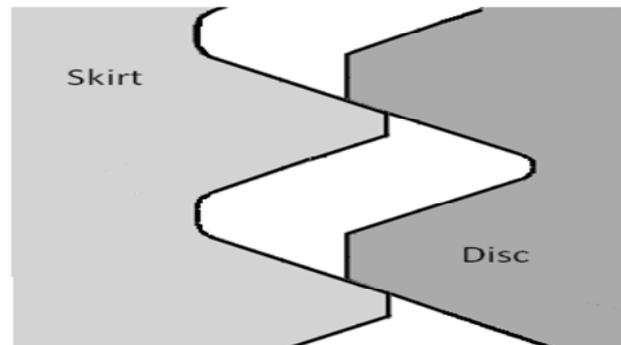


Root Cause – Disc Separation Forensics (Disc/Skirt Threads)

As-Designed Thread



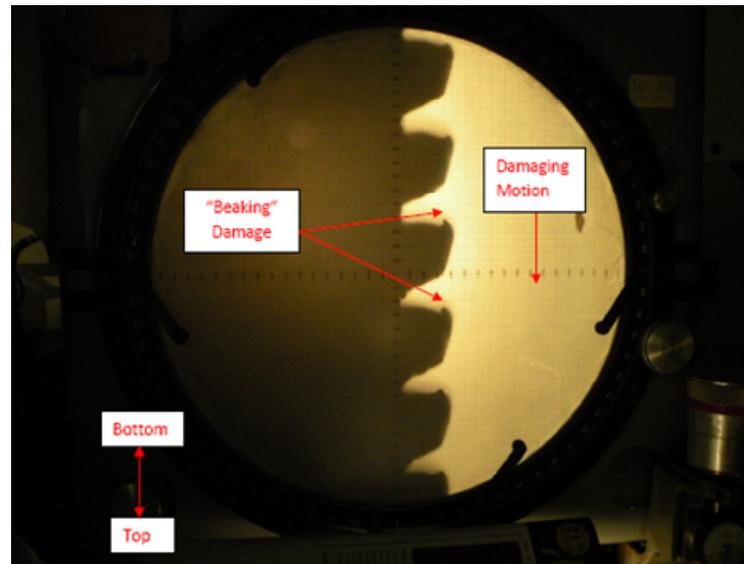
As-Found Thread





Root Cause – Disc Separation Forensics (*continued*)

Forensic examination found axial damage on the threaded connection between disc and skirt



Optical Comparator Image of Damaged Threads



Root Cause Results

- The manufacturer supplied an undersized skirt to disc connection male thread diameter under a 10 CFR 50, Appendix B program (Part 21 report submitted via revision to Licensee Event Report on April 1, 2011)
- The valve was purchased as an assembly that would not be taken apart to perform receipt inspections
 - Undersized skirt thread diameter caused the threaded connection between disc and skirt to be 38 percent of design strength
 - Pressure on skirt/stem side of disc due to downstream check valve leakage and surveillance testing configuration
 - Tack welds designed to prevent rotational, not axial separation
 - Stem and skirt pulled away from disc in open direction
- Disc was initially separated from stem/skirt before November 2008, based on Motor-Operated Valve Actuator Testing (MOVAT) data review and forensic examination
 - This indicates that the valve passed normal shutdown cooling flow in a separated condition in March 2009

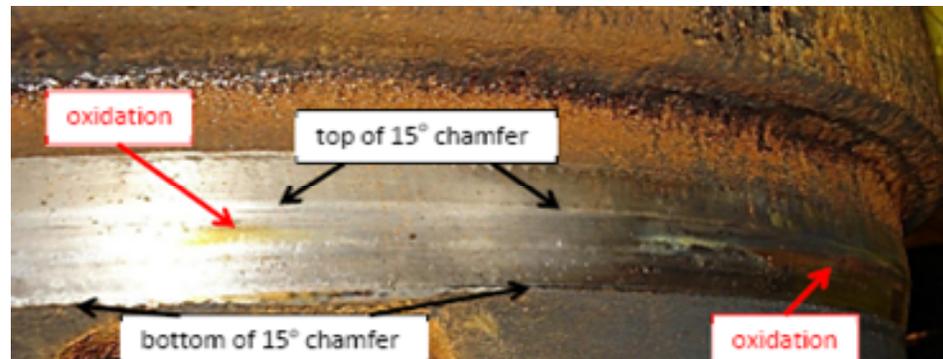
Root Cause Summary

- NRC Inspection Report 2010-05 noted “. . . the licensee’s failure to establish adequate design control and perform adequate maintenance on the Unit 1 outboard LPCI injection 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 a performance deficiency.”
- Root cause analysis determined that no licensee performance deficiency existed
 - No reasonable basis existed to examine threads and identify the undersized thread condition
- Corrective actions discussed later in this presentation
- No other root or contributing cause was identified
- We will show that the valve, while being degraded, would have performed its fire safe shutdown safety function

Functionality Analysis

- Industry research shows that pump-induced vibrations dramatically reduce frictional forces
- Idaho National Laboratory research shows static coefficient of friction behavior for stellite valve seating surfaces

74-66 valve disc contacting surface immediately after removal (11/2/10)





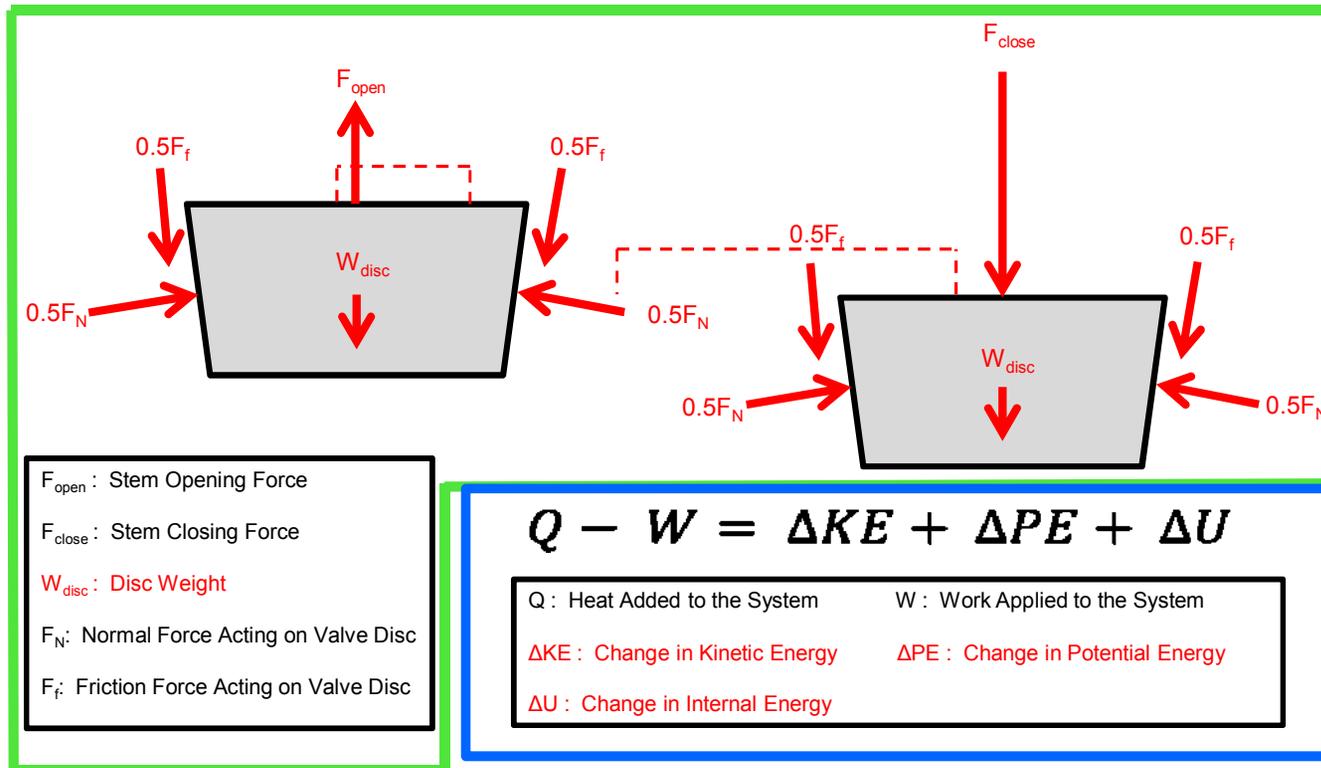
Analysis of Force Balance and Coefficient of Friction Significance

- A two-dimensional static analysis was performed to determine the normal force and coefficient of friction
- The calculated coefficient of friction was well aligned with the method discussed in Idaho National Laboratory stellite aging research¹
- An energy balance approach was used to determine the energy applied by the disc to the valve body and associated deflection from each stem stroke

¹Idaho National Engineering and Environmental Laboratory Document, INEEL/EXT-02-01021, "Results of NRC-Sponsored Stellite 6 Aging and Friction Testing," October 2002.

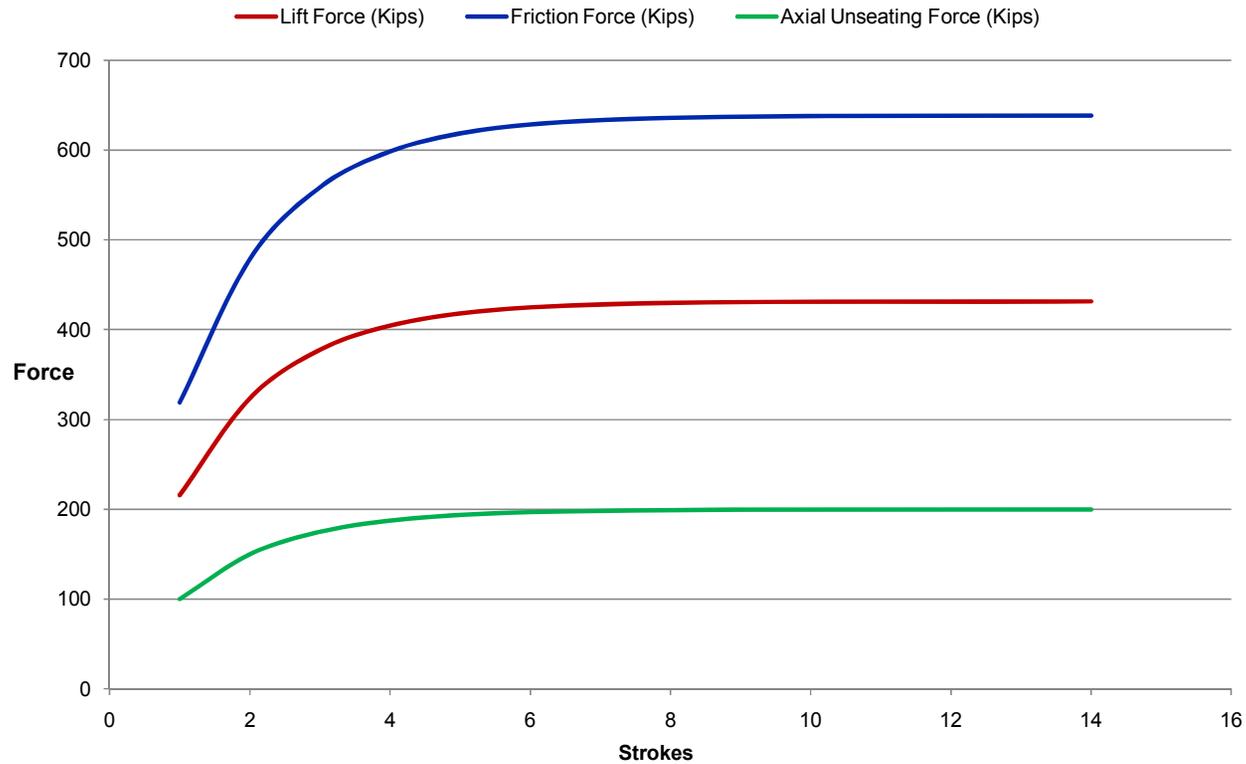


Analysis of Force Balance and Coefficient of Friction Significance *(continued)*





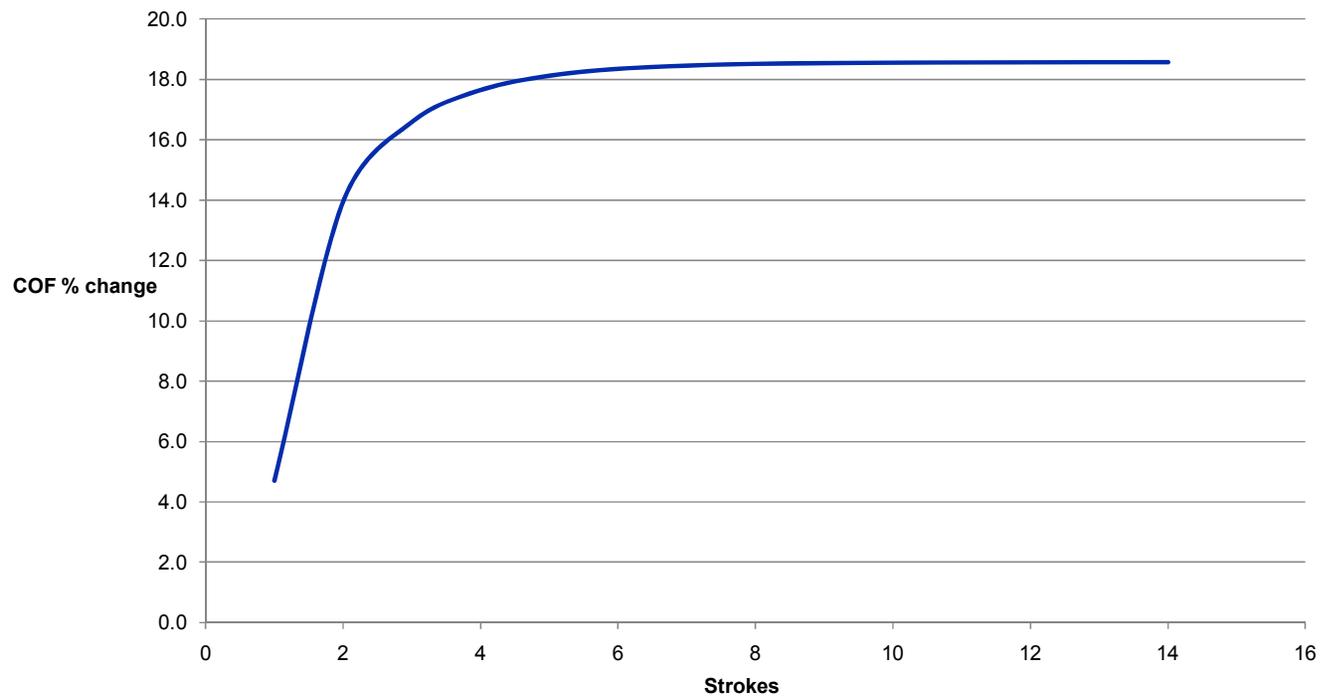
Analysis of Force Balance and Coefficient of Friction Significance *(continued)*





Analysis of Force Balance and Coefficient of Friction Significance *(continued)*

Appendix R (limiting scenario) Percent Friction Change Needed





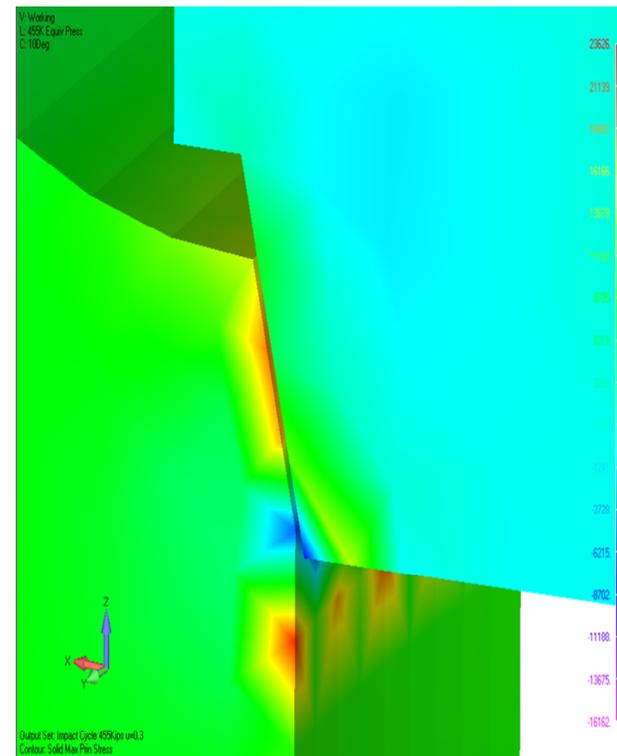
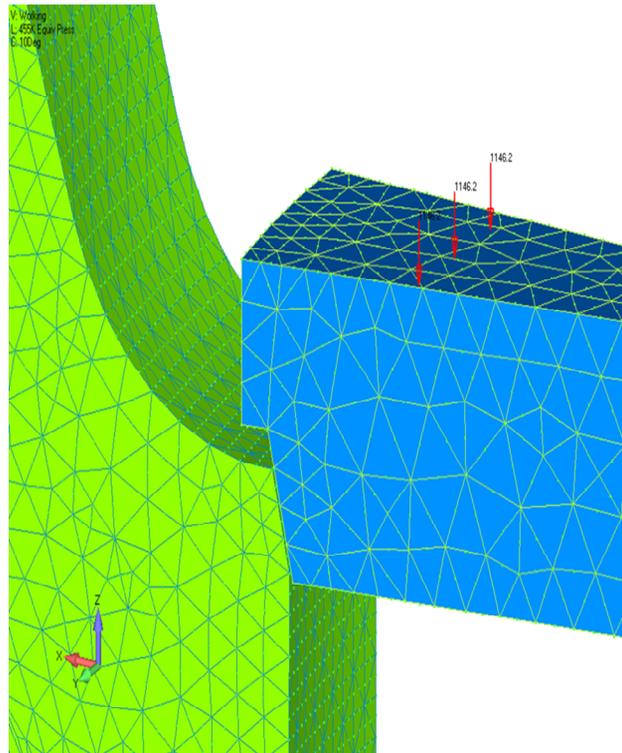
Analysis of Force Balance and Coefficient of Friction Significance *(continued)*

A finite element analysis showed:

- Close correlation with the simplified Roark stiffness used in the work energy approach
- Slight plastic deformation limited the axial deflection suggesting the simplified linear elastic approach produces a conservative frictional force



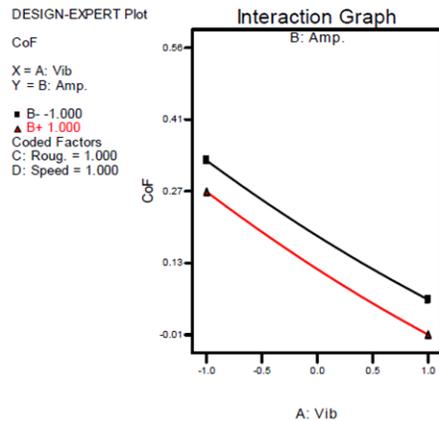
Analysis of Force Balance and Coefficient of Friction Significance *(continued)*





Vibration Effect on Coefficient of Friction Theory

- A very similar experiment conducted by researchers concludes that vibrations greatly reduce the coefficient of friction.
- The coefficient of friction is most reduced by vibration frequency and amplitude, surface roughness, speed, and quadratic terms of the surface roughness and speed.



Steel C1020 with a surface roughness and speed of 2.5 μm and 1.2 m/s; respectively

Image taken from "The Effect if Frequency and Amplitude of Vibration on the Coefficient of Friction for Metal" by Jamil Abdo and Mahmoud Tahat, Issue 7, Volume 3, July 2008. ISSN 1991-8747.

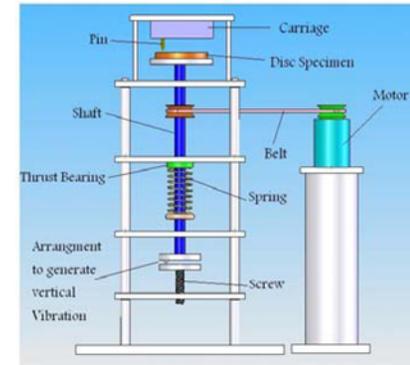


Image taken from "The Effect if Frequency and Amplitude of Vibration on the Coefficient of Friction for Metals" by Jamil Abdo and Mahmoud Tahat, Issue 7, Volume 3, July 2008. ISSN 1991-8747.

The coefficient of friction is predicted by

$$\mu = 0.313 - 0.142A - 0.0356B - 0.0458C - 0.1210D - 0.0822C^2 + 0.0561D^2$$

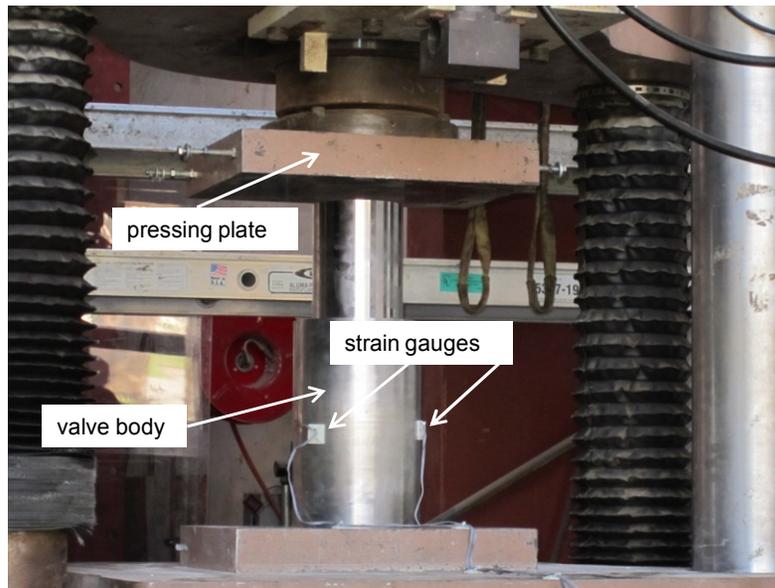
where

μ : Coefficient of Friction A: Vibration Frequency B: Vibration Amplitude
C: Surface Roughness D: Speed



Vibration Effect on Coefficient of Friction Testing

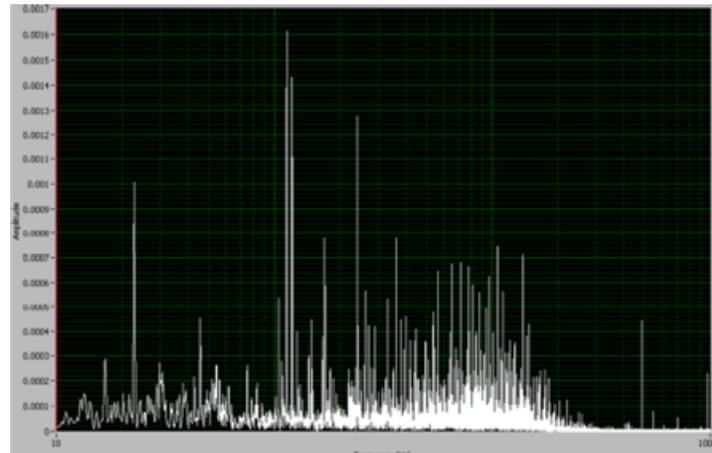
- The valve disc was compressed into the valve body by a hydraulic press.
- Strain gauges were positioned on the outside of the valve body.
- Vibrations were applied to the disc modeled on the plant configuration.





Vibration Effect on Coefficient of Friction Testing (continued)

- Pressure amplitude and frequency data were measured at Browns Ferry Nuclear Plant using a high-speed recorder with the RHR pump running
- Fast-Fourier Transform was performed and utilized in laboratory mockup testing





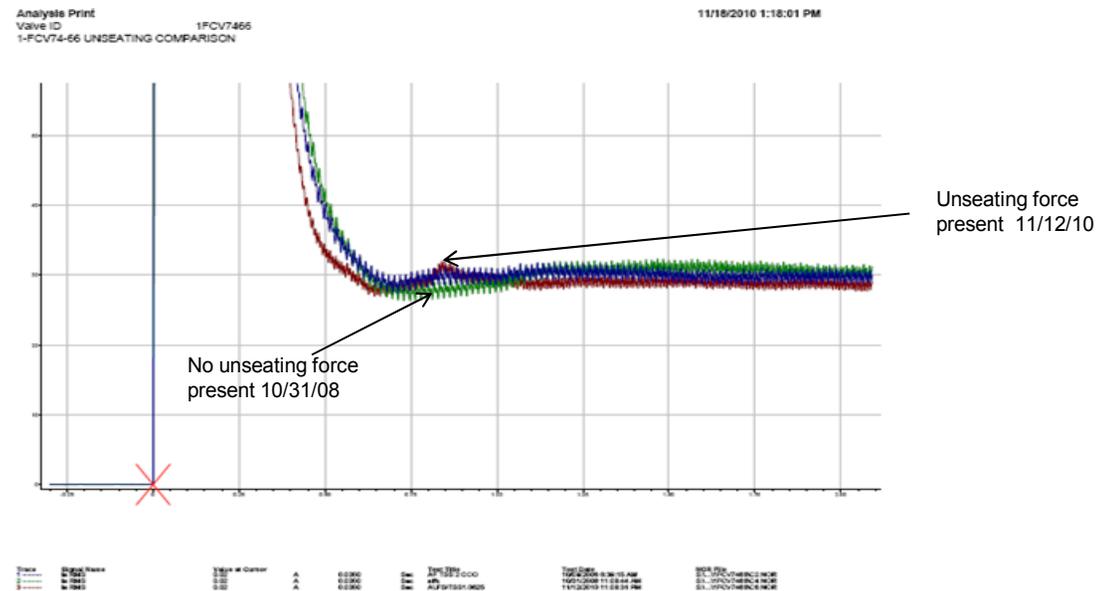
Vibration Effect on Coefficient of Friction Testing (continued)

- Multiple laboratory mockup tests concluded:
 - During multiple valve stroke surveillances, the free end of the stem “hammered” the disc into the seat
 - Disc loosens promptly with seats in clean unoxidized condition
 - Disc loosens within seven minutes with seating surfaces in roughened condition
 - System differential pressure would lift the disc, allowing proper flow as required by Safe Shutdown Analysis (SSA) for Appendix R fire (highest risk event)



Vibration Effect on Coefficient of Friction Testing (continued)

Review of MOVAT testing data, combined with stellite aging research, strongly indicates that the disc separated prior to November 2008



Note: Blue trace is 2006 data for newly lapped seating surfaces – no aged stellite.



Vibration Effect on Coefficient of Friction Testing (continued)

- Analysis concludes that four impacts of the separated stem into the disc accomplishes 94 percent of the maximum possible unseating force
- Review of plant data shows that a minimum of four impacts occurred before March 2009 when the disc lifted from the seat (operated as a check valve)
- MOVAT data (supported by stellite aging research, forensics, and the fact that unseating trace is evident following repair) indicates that the disc separated prior to November 2008
- This is strong supporting evidence that the valve loosened and operated as a check valve in March 2009



Root Cause/Engineering Analyses Conclusions

- Root cause of separation was clearly the undersized disc to skirt threads
 - This was a manufacturing deficiency and has been reported under 10 CFR 21
 - No reasonable basis existed to examine threads and identify the undersized thread condition
 - No other root or contributing cause was identified
 - There was no licensee performance deficiency
- The disc would have released and provided proper flow within seven minutes, fully supporting the limiting Appendix R fire event
 - Industry experts performed extensive analysis and laboratory testing
 - *“As a result, PII has **very high** confidence in the credibility of its findings.”¹*
 - This confidence is supported by the conclusion that valve was functional in March 2009, even though the disc was separated from the stem

¹Performance Improvement International Report, “TVA Browns Ferry Nuclear Plant, Analysis of the October 23, 2010, BFN-1-FCV-076-066 Shutdown Cooling Event,” dated March 22, 2011.

Significance Determination

TVA performed significance determination using Inspection Manual Chapter 0609, Appendix M versus Appendix F

- Appendix F does not allow quantification of defense-in-depth features
- NRC significance determination using Appendix F dominated by fire probabilistic risk assessment assumptions and conservatisms
 - Recognized by industry as overestimating baseline risk
 - Calculated fire risk conservative by factor of 5 to 10, or higher
 - Results do not conform with operating experience
- Associated RHR Loop II would have been able to fulfill fire safe shutdown function

Conclusion

- Appendix M methodology is appropriate for evaluating risk associated with failure of valve 1-FCV-74-66
- We will show that the results of Significance Determination Process evaluation should conclude that this situation is of “Very Low Safety Significance”



Significance Determination *(continued)*

Defense-in-depth associated with fire protection and fire safe shutdown

- Administrative controls to prevent fires
- Fire Protection Systems and features (including walkdowns and fire watches) to detect rapidly, control, and extinguish promptly any fires
 - Fire detection
 - Fire suppression
 - Fire barriers between fire areas
 - Dedicated onsite fire department
 - Weekly fire operations walkdowns
 - Hourly roving fire watches
 - Normal personnel traffic



Significance Determination *(continued)*

Ability of valve 1-FCV-74-66 to fulfill fire safe shutdown function

- Based on results of testing and analyses
- Results indicate valve disc freed within seven minutes
 - Would perform as check valve
 - Injection flow would be established
- Passing flow within seven-minute time period fully complies with 10 CFR 50 Appendix R SSIs and the SSA
- Operators would continue to run RHR pump to establish flow during an Appendix R event
 - Consistent with SSIs caution note to prevent exceeding pump design temperature limits

Significance Determination *(continued)*

Alternate flow paths available to support fire safe shutdown if valve 1-FCV-74-66 failed to pass flow

- Makeup to support fire safe shutdown (not specified in SSIs)
 - Condensate System (except for Turbine Building fire areas)
 - Core Spray System
 - High Pressure Coolant Injection System and/or Reactor Core Isolation Cooling System
- Long-term decay heat removal available
 - RHR Suppression Pool Cooling

Significance Determination *(continued)*

Defense-in-depth associated with design basis accidents if valve 1-FCV-74-66 failed to pass flow

- Remaining Emergency Core Cooling System subsystems available
 - LPCI associated with RHR Loop I
 - Two Core Spray Subsystems
 - High Pressure Coolant Injection System
 - Automatic Depressurization System
- Long-term decay heat removal available
 - RHR Suppression Pool Cooling

Significance determination shows that, regardless of whether valve 1-FCV-74-66 is assumed to pass flow or not, this condition was of “Very Low Safety Significance”

Performance Deficiency

Root cause of valve failure was manufacturing defect

- Preliminary cause of valve failure, identified as performance deficiency, was subsequently determined to not be the root cause
- Original manufacturer's design requirements not met
 - Undersized disc skirt threads at disc connection
- Disc skirt part of original valve assembly installed during construction in 1968-69 timeframe
- No receipt inspection of a valve assembly of this nature and classification required
 - Manufacturer provided certification documentation

Performance Deficiency *(continued)*

- Reviewed valve maintenance history
 - Valve skirt part of original valve assembly and not replaced prior to failure
 - No work performed that required measuring/confirming disc skirt thread size
- Cause is a manufacturing defect
 - Not reasonably within TVA ability to foresee and correct to prevent valve failure
- Condition should not be considered a licensee performance deficiency



Corrective Actions

- Short-term corrective actions
 - Repaired valve 1-FCV-74-66
 - Verified discs attached in all like valves, with tack welds intact and in good condition
 - Implemented controls limiting back-pressure on valves
- Long-term corrective actions
 - Restore or repair valve skirts to address potential undersized thread issue

Long-Term Fire Strategies

- Operator Manual Action (OMA) reduction
- SSI revisions
- Proactive installation of NFPA 805 transition modifications
- Driving down risk impacts utilizing NRC risk methodology



SSI Revision Goals

- Reduce plant risk in serious fire events
 - Reduce instances of SISBO actions
- Add branching steps to SSIs
 - Entry conditions would remain unchanged
 - Operator would be directed to use alternate safe shutdown methods if the SSI cannot be executed
 - Alternate paths include Condensate System, Core Spray System, High Pressure Coolant Injection System, and/or Reactor Core Isolation Cooling System



SSI Revision Goals *(continued)*

- Address Appendix R compliance
 - Reduce (but not eliminate) number of OMAs
- Reduce complexity of SSIs
- Support of NFPA 805 implementation
 - Implement post-transition shutdown strategies and procedures in advance
 - Implement modifications proactively

SSI Revision Plan

Phase I – Turbine Building and Intake

- Complete most risk sensitive area (planned for July 2011)
- Use upgraded SSA currently in progress for NFPA 805 transition
- Risk map (using conservative NRC approach) shows significant OMA issue risk reduction in July 2011
- Independent team established to execute this in parallel with NFPA 805 transition
- Utilizes plant recently completed modifications
 - Turbine Building/Intake Structure fire barrier
 - Cable tray covers
 - Incipient detection



SSI Revision Plan *(continued)*

Phase I – Turbine Building Specifics

- New fire barrier completed
 - Separate Turbine Building from intake
 - 3-hour rated
 - Allows a separate SSA for the Turbine Building
 - One train free of fire damage
- Symptom-based procedure
 - Essentially eliminates SISBO for this fire area
 - Allows use of available equipment
 - Additional precautions and instructions specific to the fire area
 - Protection of the credited train



SSI Revision Plan (Current Schedule) (continued)

First Phase (Turbine Building Fire Area Separation: 25 and 26)



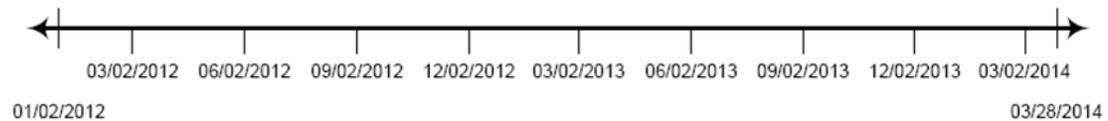
04/04/2011 07/29/2011

Second Phase (Fire Areas with CDF > 1E-6: 5, 6, 3-3, 3-4, 1-5, 2-3, 9, 12, 22, 23)



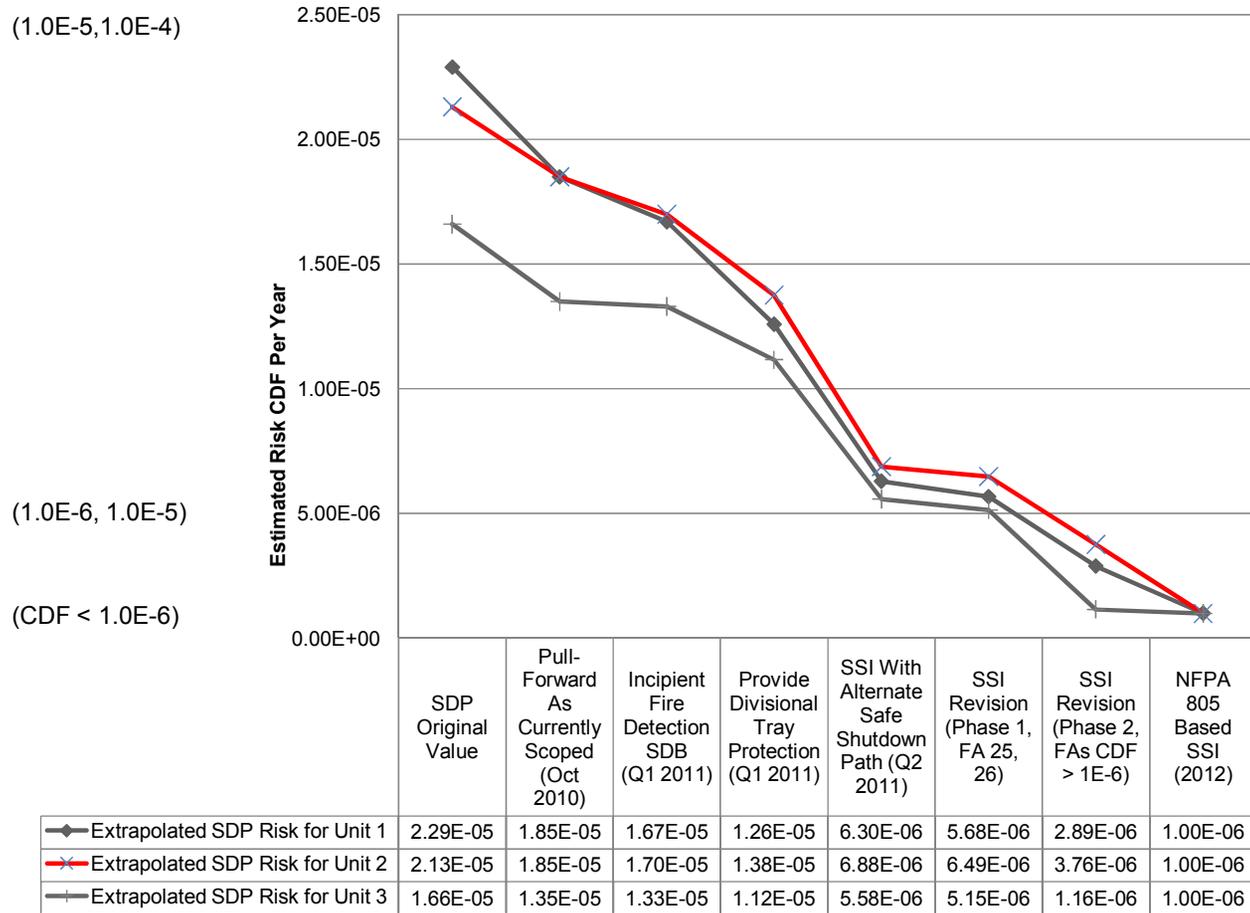
08/08/2011 12/30/2011

Third Phase of SSI Revision (All Fire Areas Between LAR and 6 months after SER Date, Assume 18 month NRC review)





Extrapolated SDP Risk Estimates (Current Schedule)



SSI Revision Plan *(continued)*

Phases II and III – Additional High Risk Fire Areas

- Utilize advantages gained from upgraded SSA
- Improve shutdown strategy and procedures, if possible under Appendix R deterministic requirements
- Plant modifications, if feasible
- Will require completion of cable routing and analysis for affected areas



SSI Revision Team Approach

- Team managed from TVA corporate offices with work being performed in Chattanooga and vendor locations
- Team composition
 - Browns Ferry Nuclear Plant operations procedure writer and operations trainer
 - TVA corporate engineering manager for technical direction and oversight
 - Vendor engineers (2) for SSA support
 - Vendor engineers (3) for engineering design change development and support

Risk Reduction Summary

- TVA remains firmly committed to NFPA 805 transition
- Strong management support and oversight
 - Detailed transition schedule
 - Extensive resources onsite and at corporate office
 - Regular progress meetings with senior leadership
- Modifications and safe shutdown procedure improvements are continuing aggressively
- High focus is maintained on fire protection system health

Closing Remarks

TVA has shown conclusively that:

- There was no licensee performance deficiency
- LPCI valve 1-FCV-74-66 would have performed its safety function under Appendix R fire conditions
- Significance determination results in a finding of “Very Low Safety Significance”

TVA has also shown how accelerating some actions from the transition to NFPA 805 will quickly reduce fire risk at the Browns Ferry Nuclear Plant

- Changing SSIs to allow use of alternate safe shutdown paths
- Reducing the number of times that SISBO actions are taken
- Completed plant modifications such as incipient detection and cable tray covers