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

November 28, 2011

Mr. T. Preston Gillespie, Jr.
Site Vice President
Duke Energy Carolinas, LLC
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: PUBLIC MEETING SUMMARY – OCONEE NUCLEAR STATION – DOCKET NOS. 50-269, 50-270 AND 50-287

Dear Mr. Gillespie:

This refers to the Category 1 public meeting which was held on November 16, 2011, in Atlanta, GA. The purpose of this meeting was to discuss the safety significance of two preliminary greater than Green findings with two associated Apparent Violations (AV) that were documented in NRC Inspection Report 05000269,05000270, 05000287/2011018 (ML11277A253). A listing of meeting attendees and information presented during the meeting are enclosed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room (PDR) 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) 997-4607.

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

Enclosures:
1. List of Attendees
2. NRC PowerPoint Presentation
3. Licensee PowerPoint Presentation

cc w/encls: (See page 2)

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

ADAMS: ☐ Yes ACCESSION NUMBER:_ ML11333A342_ ☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII:DRP						
SIGNATURE	/RA/						
NAME	JBartley						
DATE	11/28/2011						
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: DOCUMENT4 cc w/encls:

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Letter to T. Preston Gillespie, Jr. from Jonathan H. Bartley dated November 28, 2011

SUBJECT: PUBLIC MEETING SUMMARY – OCONEE NUCLEAR STATION – DOCKET NOS. 50-269, 50-270 AND 50-287

Distribution w/encls:
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OCONEE NUCLEAR STATION - REGULATORY CONFERENCE
November 16, 2011

NAME	COMPANY AND TITLE
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Tracy Saville	Duke Energy, NGO Nuc. Manager
Jim Hammer	DUKE ENERGY, ONS, TES MANAGER
Bryan Carroll	Duke Energy NGO PRA Manager
Lee Kanipe	Duke Energy NGO Sr. PRA Engineer
John Klingenfus	AREVA, Advisory Eng LOCA Methods
Eric Henshaw	Duke Energy NGO Safety Analysis Principal Eng.
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Scott Batson	Duke Energy, Station Manager
Greg Byers	Duke Energy, T-H Analyst
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Mike Kitlan	Duke Energy, SA Analyst
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Victor McCree	NRC, Region II Regional Administrator
Rick Croteau	NRC Region II Div. of Reactor Projects Director
Jon Bartley	NRC, Region II, Chief Branch 1 Division of Reactor Projects
Andy Sabisch	NRC Region II, Senior Resident Inspector Oconee Nuclear Station
Geoff Ottenberg	NRC, Region II, Resident Inspector Oconee Nuclear Station

Enclosure 1

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OCONEE NUCLEAR STATION - REGULATORY CONFERENCE
November 16, 2011

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Anna Mitchell	Anderson Independent

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

OCONEE REGULATORY CONFERENCE

November 16, 2011

Enclosure 2

Agenda

- OPENING REMARKS AND INTRODUCTION
- NRC REGULATORY AND ENFORCEMENT POLICY
- STATEMENT OF ISSUES AND APPARENT VIOLATIONS
- DUKE ENERGY CAROLINAS RESPONSE
- BREAK/NRC CAUCUS
- NRC FOLLOW UP QUESTIONS
- CLOSING REMARKS
- PUBLIC QUESTIONS

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Protecting People and the Environment

OCONEE REGULATORY CONFERENCE

November 16, 2011

Issue #1

A licensee-identified preliminary Greater Than Green Apparent Violation of 10 CFR 50, Appendix B, Criterion III, Design Control, was identified when the licensee failed to maintain design control of the Standby Shutdown Facility. The SSF pressurizer heater breakers and associated electrical components were not maintained as safety-related components nor seismically qualified as specified in the SSF licensing basis documents. Because the safety significance of this finding is preliminary Greater Than Green, it is being treated as an NRC-identified finding. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective in that failure to maintain equipment qualification did not provide reasonable assurance that the Standby Shutdown Facility Auxiliary Service Water subsystem would perform its safety function.

10 CFR Part 50, Appendix B, Criteria III, Design Control, required, in part, that measures shall be established to assure that deviations from appropriate quality and design standards are controlled and that the review for suitability of application of equipment essential to safety-related functions of structures, systems, and components is maintained. Technical Specification 3.10.1.A required that, with the Standby Shutdown Facility Auxiliary Service Water inoperable, the system shall be restored to an operable status within seven days or the unit placed in Mode 3 within 12 hours and Mode 4 within 84 hours.

Contrary to the above, from 1983 until June 1, 2011, the licensee failed to review for suitability of application of equipment essential to safety-related functions of structures, systems, and components. The licensee failed to maintain the SSF pressurizer heater breakers and associated electrical components as safety-related QA-1 and seismically-qualified components in accordance with the licensing and design bases. As a result, the Standby Shutdown Facility was inoperable from 1983 until June 1, 2011, a period in excess of Technical Specification 3.10.1.A allowed outage time.

The apparent violations discussed in this regulatory conference are subject to further review and are subject to change prior to any resulting enforcement action.

Issue #2

An NRC-identified preliminary Greater Than Green Apparent Violation of 10 CFR 50, Appendix B, Criterion III, Design Control, was identified for the licensee's failure to install Standby Shutdown Facility pressurizer heater breakers that were qualified for expected environmental conditions inside of containment during design basis events. The licensee installed replacement breakers and the Standby Shutdown Facility was declared operable without testing to support that the replacement breakers would function under elevated containment temperatures. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective in that failure to maintain equipment qualification did not provide reasonable assurance that the Standby Shutdown Facility Auxiliary Service Water subsystem would perform its safety function. This finding had a cross-cutting aspect in the area of Human Performance under the Procedural Compliance aspect of the Work Practices component in that the licensee failed to follow the requirements set forth in EDM 601, Engineering Change [H.4(b)].

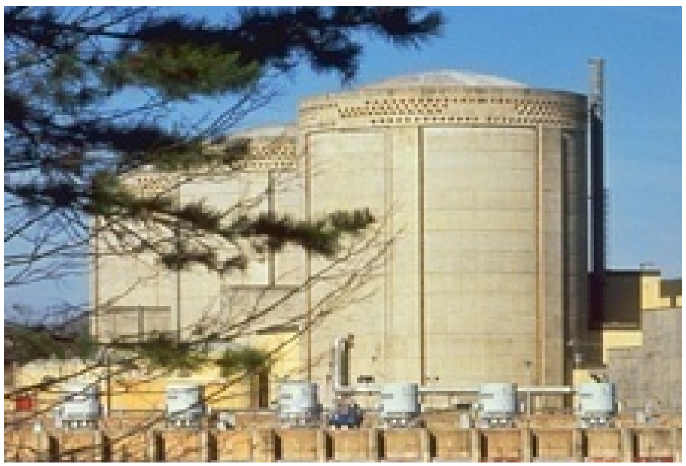
10 CFR Part 50, Appendix B, Criteria III, Design Control, required, in part, that measures shall be established to assure that deviations from appropriate quality and design standards are controlled and that the review for suitability of application of equipment essential to safety-related functions of structures, systems, and components is maintained. Procedure EDM 601, Engineering Change, Section 601.5.2.1, Engineering Change Program – Design Phase, Commercial Controls, required that components acquired commercially be tested to verify they will perform the required safety function prior to actual installation. Technical Specification 3.10.1.A required that, with the Standby Shutdown Facility Auxiliary Service Water inoperable, the system shall be restored to an operable status within seven days or the unit placed in Mode 3 within 12 hours and Mode 4 within 84 hours.

Contrary to the above, on June 4, 2011, a review for suitability of application of equipment essential to safety-related functions of structures, systems, and components was not performed. The licensee developed a modification that installed replacement breakers in the circuit supplying power to the pressurizer heaters from the Standby Shutdown Facility without test data to demonstrate that they would function at elevated containment temperatures and maintain the Standby Shutdown Facility functionality in accordance with the licensing and design bases. As a result, the Standby Shutdown Facility was inoperable from the time of new breaker installation (Unit 1 on June 8, 2011; Unit 2 on June 6, 2011; and Unit 3 on June 7, 2011) until August 20, 2011, a period in excess of Technical Specification 3.10.1.A allowed outage time.

Public Questions

Enclosure 2

Oconee Nuclear Station Regulatory Conference



Standby Shutdown Facility (SSF) Pressurizer Heater Breakers Units 1, 2, & 3

NRC Region II
Atlanta, Georgia
November 16, 2011

Enclosure 3

Duke Participants

Preston Gillespie	Oconee Site Vice President
Tom Ray	Oconee Engineering Manager
Scott Batson	Oconee Station Manager
Tracy Saville	NGO Nuclear Engineering Manager
Terry Patterson	Oconee Safety Assurance Manager
Bob Guy	Oconee Organizational Effectiveness Manager
Bill Pitesa	Senior Vice President, Nuclear Operations

Agenda

Opening Remarks	P. Gillespie
Original Design Configuration	T. Ray
Timeline, Breaker Selection & Testing	T. Ray
Operations Role in the Operability Determination	S. Batson
Significance Determination	T. Saville
Regulatory Perspectives	T. Patterson
Cause Analysis and Corrective Actions	B. Guy
Closing Remarks	B. Pitesa

Opening Remarks

Preston Gillespie
Oconee Site Vice President

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Opening Remarks

- NRC proposed two findings related to the PZR HTRs powered from the SSF:
  - A legacy issue with the original design configuration
  - A current performance issue related to the breakers installed and declared operable
- Oconee agrees with NRC's characterization of the:
  - Apparent violation of Appendix B, Criteria III (legacy issue)
  - Apparent violation of Appendix B, Criteria III (current performance issue)
- Information will be presented to address:
  - The self-identification of the SSF PZR HTR BKR problem in the original design
  - The immediate actions taken and actions taken to prevent recurrence
  - Assessment of the SSF design and licensing bases
  - The legacy issue alignment with the "Old Design Issue" criteria in IMC 0305

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Opening Remarks

- New or additional information is included to address the following:
  - New test results for replacement breakers under a more representative environment
  - Plant response to the loss of PZR HTRs in an SSF event
    - Thermal-hydraulic modeling & confirming calculations
    - Simulator response consistent with analytical predictions
- Key drivers contained in the probabilistic risk assessment:
  - Human error probability
  - Bus duct fire analysis
  - Cable failure analysis
  - PSV failure probability
- Oconee's determination of risk significance related to the findings are:
  - Legacy issue - low to moderate safety significance
  - Current performance issue - very low safety significance

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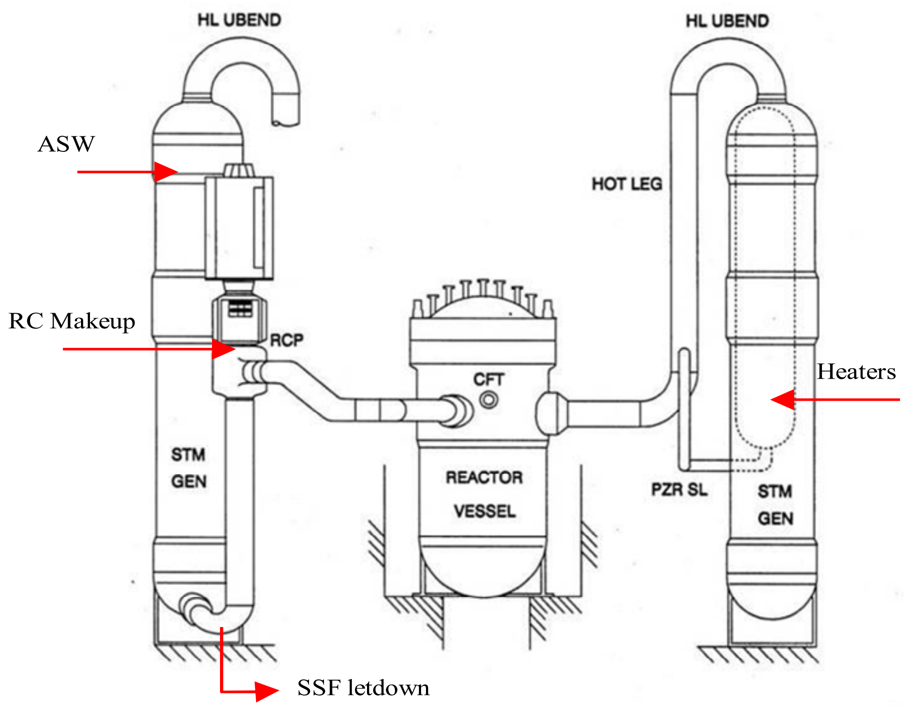
Original Design Configuration

Tom Ray
Oconee Engineer Manager

SSF Equipment



- ASW
  - Injects through OTSG upper nozzles
- RC Makeup
  - Injects through RCP seals
  - 29 gpm (nominal)
- SSF letdown line
- Pressurizer heaters
  - Subset of normal pressurizer heaters are powered from the SSF

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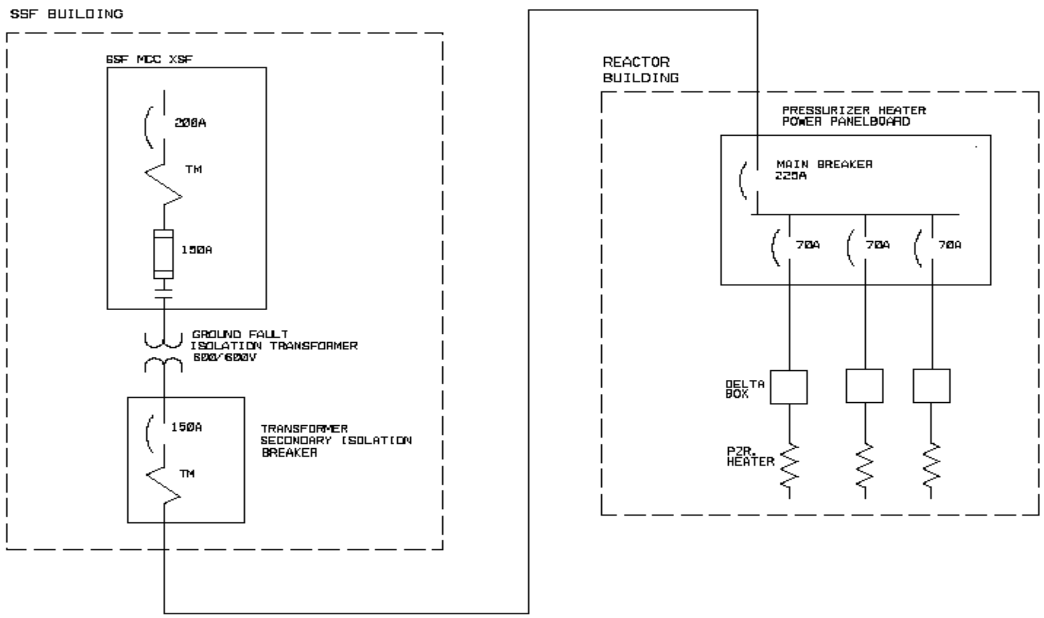
SSF PZR Heaters Schematic

Simplified Schematic of SSF PZR Heater Circuitry



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SSF PZR Heaters Design Basis

- Oconee self-identified a deficiency with the SSF PZR HTR BKRs
  - Deficiency identified during investigations associated with a future modification
  - Containment temperature increases on loss of building cooling
  - Thermal overloads are sensitive to increased ambient temperature
- Independent assessment of the SSF design & licensing bases
  - The SSF is the system with the highest risk worth at Oconee
  - The assessment team includes independent and external members
    - Contractors with previous SSFI and CDBI experience included on team
    - Assessment represents a sizable commitment of time and resources
    - Assessment outcome will improve our performance

Timeline,
Breaker Selection & Testing

Tom Ray
Oconee Engineer Manager

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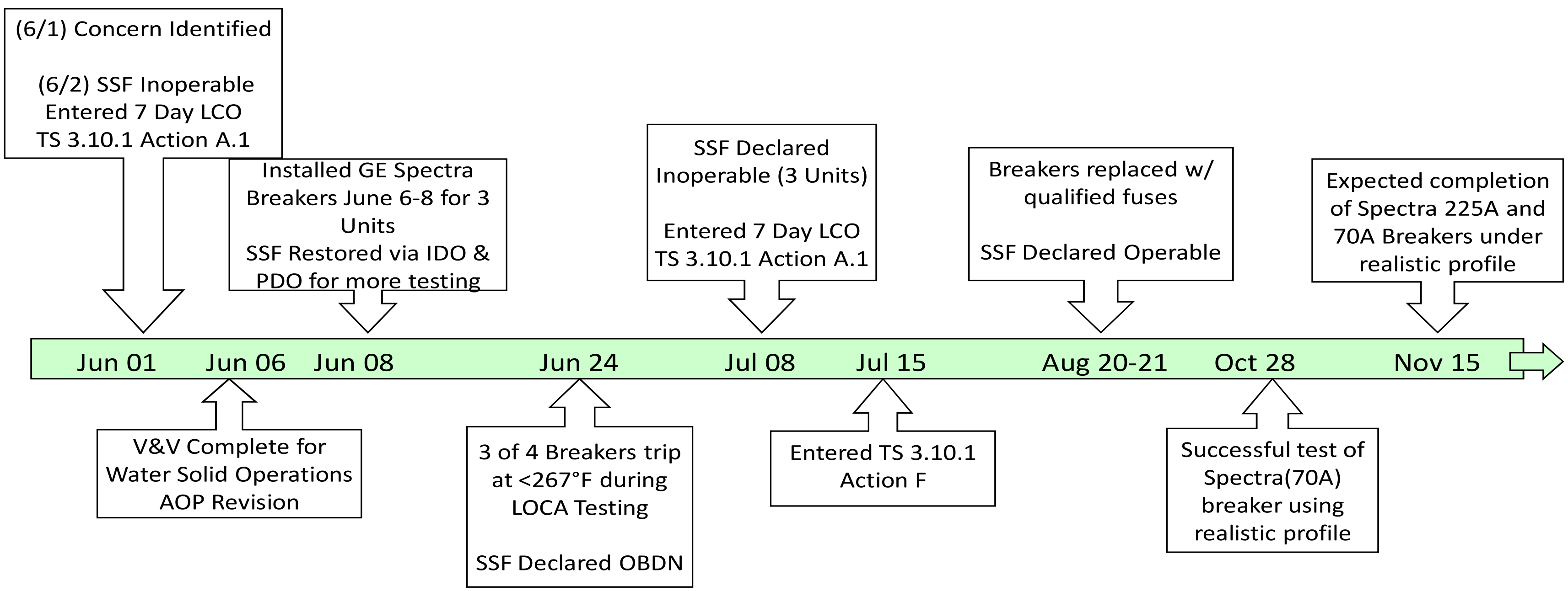
Timeline



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Breaker Selection

- Relative to BKR selection, Bounding Conditions for the SSF Function:
  - The SSF mission time is 72 hours
  - Containment temperature profile
  - Loss of containment cooling with loss of all AC is the limiting condition
  - Aging was addressed by limiting design life
  - Low exposure levels based on location and event type
- Selection of GE Spectra Breakers for Installation:
  - Trip function less sensitive to temperature
  - Breaker function and materials were reviewed for suitability
  - Testing was performed at a Duke facility to prove the BKRs would survive the high temperature ambient environment
  - Breakers were placed in service while testing activities continued

Breaker Test Results

- Testing was conducted at Kinetics in June (GE Spectra Breaker):
  - Testing was conducted in a LOCA chamber
  - Breakers were powered during the test
  - Testing profile increased temperature faster than actual conditions
  - Testing suspended when 3 of 4 failed
- Fuses were qualified at Kinectrics for use as a permanent modification
- Recent (November) testing is being conducted with the GE Spectra Breakers
  - Testing with representative event temperature profile
  - Breakers powered while in test oven at temperature (70A & 225A Breakers)
    - Received report 11/14/11 from test lab the 225A Breaker tripped
  - Breakers will be inspected and verified to trip before and after the 72 hr test

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Summary

- EDM-601, Engineering Change Process, requires design inputs to be completed prior to declaring operable
- Based on preliminary insights from current root cause effort, weaknesses in implementing the modification processes were as follows:
  - Engineers had confidence in their engineering evaluation and testing was viewed as confirmatory rather than qualifying
  - Peer reviews did not challenge the implementation of the process
- Weaknesses in implementing the operability processes
  - Decision-making influenced by previous use of water solid operations
  - Modification improved but did not ensure the design function

Operations Role in the Operability Determination

Scott Batson
Oconee Station Manager

Enclosure 3

Standby Shutdown Facility

Enclosure 3

Operability Determinations

- Immediate Determination of Operability – June 6 - 8 , 2011
  - OSM was knowledgeable of testing and results
  - Peer review was obtained on the IDO prior to acceptance based upon reasonable assurance of operability
  - Test verified temperature had no impact on trip settings
  - Breaker materials reviewed and supported the ambient insensitivity position
  - Confidence in Water Solid Operation as alternate means to meet pressure control function
  - Required PDO to validate assumptions in the IDO

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Operability Determinations

- Determination of Operability – June 24, 2011
  - Licensed Operators trained that our analysis demonstrates that RCS pressure control from the SSF with Water Solid Operations is a technically sound and effective mitigation strategy
  - Operating procedure guidance on Water Solid Operations since 2002
  - Operating crews trained on Water Solid Operations in the SSF Simulator
  - Procedures were revised in 2011 to include additional identification steps
  - Actions, indication and controls are simple and readily accessible in the SSF Control Room
  - Sufficient time is available to complete operator actions

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Insights

- Weakness in implementation of the Operability Determination Process
  - Did not challenge process / procedure requirements
  - Peer review did not challenge process / procedure requirements
  - Decision-making influenced by previous use of Water Solid Operations
  - Did not recognize the need for prior NRC approval for Water Solid Operation
- Failed to recognize Water Solid Operation as a Compensatory Measure
  - Actions were inconsistent with NSD 203, "Operability Determination Process"
  - Inappropriately credited existing procedural guidance
  - A 10 CFR 50.59 review should have been performed
- Operability Determination Process improvement project will incorporate these insights.

Significance Determination

Tracy Saville
NGO Nuclear Engineering Manager

Enclosure 3

Significance Determination Overview

- SSF Operation for Normal/Loss of Heater Conditions
- PRA Considerations for Water Solid Operations (WSO)
- Operating Procedural Flow Paths
- Key Differences with NRC
  - Human Reliability Analysis
  - Bus Duct Fire Analysis
  - Cable Failure Analysis
  - PSV failure probability
- Adjustments for Current Performance Issue
- Risk Results

Initial SSF Activation & System Response

- SSF manned and activated, diesel generator started
  - ASW provides heat sink to SGs
  - PZR heaters energized to maintain RCS pressure
  - RC makeup pump provides RCP seal cooling/RCS makeup
- ASW flow controlled to RCS pressure setpoint
  - RCS slowly cools to offset RC makeup
- RCS trends to stable hot standby condition
  - T-cold ~550F (controlled by Main Steam relief valves)
  - Natural circulation SG water levels
  - Stable RCS pressure & PZR level
  - SSF letdown is cycled to control PZR level

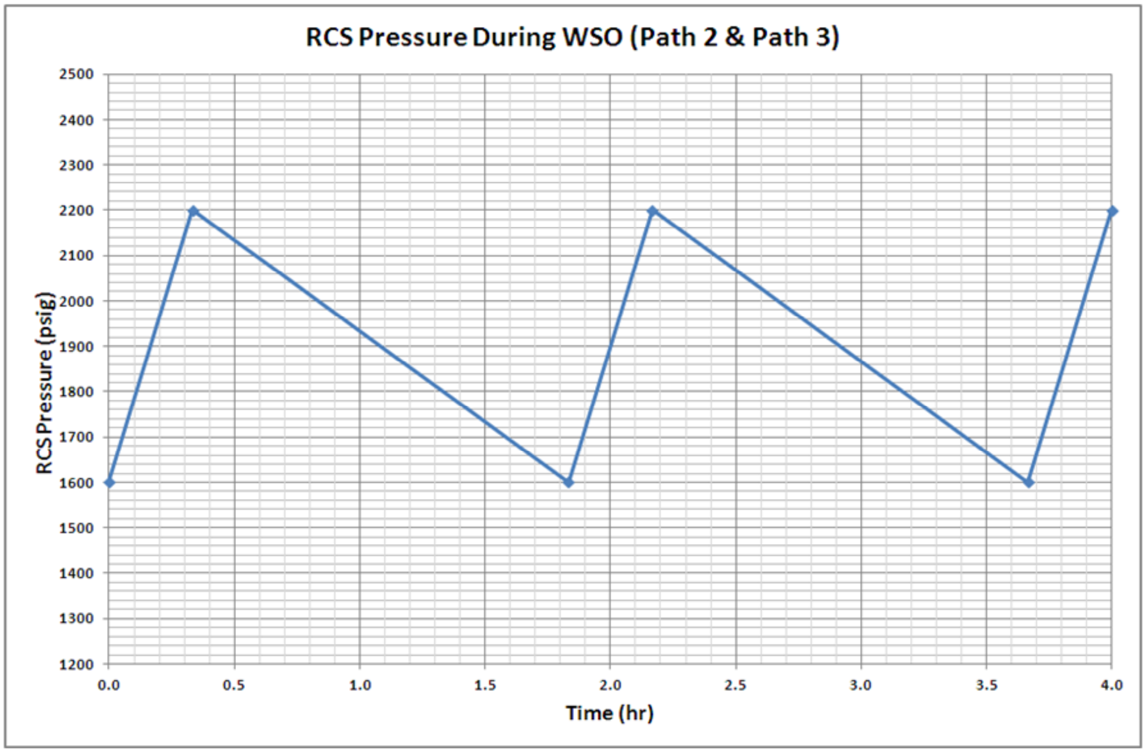
Response to Loss of Heaters

- Plant can be controlled via three procedural flow paths
  1. Throttle ASW to control to RCS pressure (1950-2250 psig) with letdown line initially open
  2. Throttle ASW to maintain SG level setpoint and cycle letdown line block valve to maintain RCS pressure 1600-2200 psig (most probable strategy used)
  3. Hybrid, begins with path 1, transitions to path 2 once effect of open letdown line is recognized
- Strategy used is dependent upon conditions at time of heater loss and recognition of the symptom

Operator Actions in WSO

- Procedural guidance and operator action times are sufficient to maintain RCS pressure within prescribed limits of 1600-2200 psig
- T/H calculations show: (Path 2 & Path 3)
  - Pressure increase 1600 - 2200 psig > 20 min
  - Pressure increase 2200 - 2500 psig > 10 min
  - Pressure decrease 2200 - 1600 psig > 90 min
- Plant simulator used to validate operator's ability to control the plant in WSO

RCS Pressure Trace - WSO



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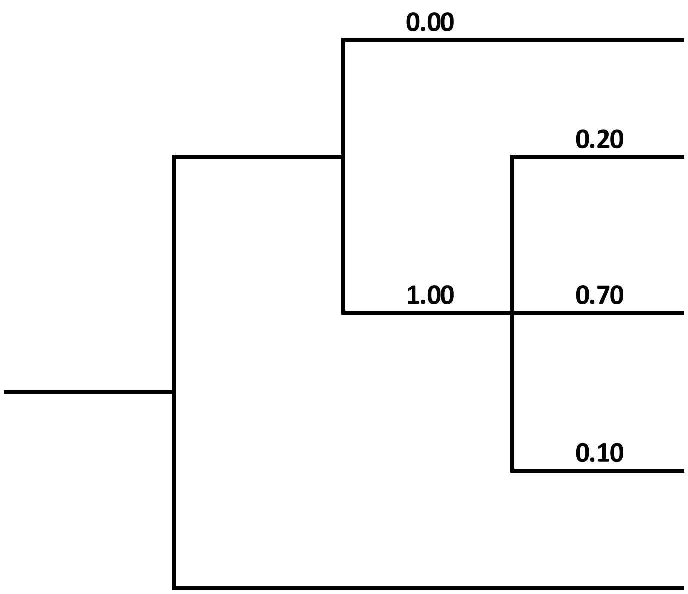
PRA Considerations for WSO

- WSO is a viable means of maintaining the RCS in a safe and stable condition
  - Based on extensive Duke analysis which has been provided to NRC
- Risk significance of WSO following total heater loss is low and involves:
  - Operator failure to control RCS pressure within prescribed procedural limits
  - Component failure of letdown line block valve

Procedural Flow Path Event Tree

SSF Event	SSF Successful	Heaters Successful	Procedure Path	Comments



Base Case - No delta risk

Path 1 - Throttle ASW to control to RCS pressure (1950-2250 psi) with letdown line initially open

Path 2 - Throttle ASW to maintain SG level setpoint and cycle letdown line block valve to maintain RCS pressure 1600-2200 psi

Path 3 - Begins like path 1 and transitions to path 2 once effect of open letdown line is recognized

Base Case - No delta risk

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Key Differences

- Human Reliability Analysis (HRA)
  - Detailed modeling of each procedural flow path
  - Duke uses an accepted industry analysis approach employed in the EPRI HRA Calculator Tool
  - For Path 2, the combined Human Error Probability is approximately 0.018.
  - NRC used a screening value of 0.1
  - A factor of 5.5 difference in results
- Duke applied a probability of 0.1 for the PSV failure
  - This is same probability the NRC used in previous ONS SDP
  - NRC used a screening value of 1.0

Key Differences

- Bus Duct Fire Initiating Event Frequency
  - Duke's analysis of record used a plant specific value of 1.2E-03 based on an EPRI data analysis report
  - The NRC Phase III analysis used a generic frequency of 3.3E-03 from an NFPA-805 FAQ
  - The NRC used a value of 2.4E-03 in a final significance determination in 2010
  - Duke believes the Oconee specific Bus Duct fire initiation probability the NRC used in its previous risk analysis is a better estimate than the generic NFPA-805 FAQ Bus Duct frequency used in the screening
  - Our results reflect the use of the plant specific value docketed in the NRC final significance determination in 2010

Key Differences

- Cable Failure Analysis
  - Duke is performing a detailed cable failure analysis to more accurately determine the likelihood of a blackout.
  - Duke has preliminary results which showed:
    - A significant reduction in the Unit 3 Blackout Frequency
    - A modest reduction in the Unit 1 Blackout Frequency
    - Insignificant change in the Unit 2 Blackout Frequency
  - NRC analysis assumes that a station blackout would occur if these cable trays were affected.
  - Preliminary information has been recently transmitted. Final report will be transmitted when finalized (~Nov. 18)

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Adjustments for Current Performance Issue

- Reduced exposure period of ~75 days vs. 12 months
- Event tree limited to Path 3 based on procedure changes made on June 6
  - Has lower failure probability than Path 2
  - Fewer valve strokes and required operator actions
- Reflects lower tornado activity during exposure period (~June – August)
- Duke expects a lower breaker failure probability based on preliminary test data (sensitivity performed).

Risk Results

- Legacy Issue (using NRC's 2010 bus duct frequency):
  - Unit 1 – low E-06
  - Unit 2 – low E-06
  - Unit 3 – low E-06
- Current Performance Issue
  - Mid E-07 (assuming 100% breaker failure probability)
  - Mid E-08 (assuming 10% breaker failure probability)
- Risk associated with SSF heater failure is much lower than the NRC Phase III SDP analysis.

Regulatory Perspectives

Terry Patterson
Oconee Safety Assurance Manager

Enclosure 3

Regulatory Perspectives: Legacy Issue

- Apparent Violation of Appendix B, Criterion III, Design Control:
  - Oconee agrees with the apparent violation
  - Corrective actions have been taken to restore compliance
  - Comprehensive actions taken or planned for extent-of-condition
- Significance:
  - This issue has a low to moderate risk significance
  - Oconee has offered its risk insights regarding the PRA analysis:
    - Human Reliability Analysis
      - Water solid operations as a method of RCS pressure control
    - Cable tray interaction with bus duct fires

Old Design Issue Criteria

[IMC 0305 Section 11.05(a)]

- Criterion 1 – Licensee Identified:
  - Oconee engineers identified the potential for similar conditions during modification activities in an unrelated area
  - This behavior demonstrates a strong questioning attitude
  - The condition was promptly entered into the corrective action program
- Criterion 2 - Immediate and long-term corrective actions to prevent recurrence
  - Immediate actions were prompt and comprehensive
  - Condition was addressed for all three units
  - Action statement entered for all units within hours of identification
  - Improvements from the CAP Improvement project realized
  - Qualified fuses installed to correct condition

Old Design Issue Criteria
[IMC 0305 Section 11.05(a)]

- Criterion 3 – Opportunity for Prior Discovery:
  - The degraded condition was not self-revealing
  - The degraded condition could not be observed during normal operations, routine testing or maintenance.
  - Operating Experience review
- Criterion 4 – Current Performance:
  - The breakers with thermal overloads had been installed since original construction
  - Identification was the result of a strong questioning attitude
  - The deficient condition was promptly entered into CAP and evaluated for operability
  - Immediate actions have been taken and long term actions are in progress

Regulatory Perspectives - Current Performance Issue

- Apparent Violation of Appendix B, Criterion III, Design Control:
  - Oconee agrees with the apparent violation
  - Corrective actions have been taken to restore compliance
  - Comprehensive actions taken or planned for extent-of-condition
- Significance:
  - Reduced exposure period of ~75 days vs. 12 months
  - Procedure changes implemented for WSO
  - Conducting test of Spectra BKRs with revised temperature profile
  - This issue has a very low risk significance

Cause Analysis & Corrective Actions

Bob Guy
Organizational Effectiveness Manager

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Legacy Issue: Cause Determination

Preliminary Evaluation of Initial SSF Design Scoping 1978-1982

- Engineering review of the capabilities of re-purposed components (pressurizer heaters) to function under new conditions was not performed.
- Scoping of the project was inadequately specified and excluded the consideration of the pressurizer heater qualifications.
- Existing practice did not analyze re-purposed equipment to ensure that it would function under any new conditions.

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Legacy Issue: Corrective Actions

Extent of Condition

- Pressurizer heater panel board breakers - associated power circuitry
- Equipment breakers that must support safety equipment that will not function due to unexpected tripping due to environmental conditions
- Resolution of over 70 issues

Corrective Actions

- Design Change Processes
- Documentation and Training
- Overall Independent Assessment (IA) of the SSF
  - Two-phase approach - SSF Current Licensing Basis (CLB) Submittals and modifications

Current Performance Issue: Cause Analysis

Root Cause Evaluation

Combined team to fully assess:

- Process applied during Engineering Changes
- Engineering Changes for GE Spectra circuit breakers
- Operability Determination application
- Operability Determination process

Current Performance Issue
Corrective Actions

Safety Culture Enhancements

- CAP Improvement Project
  - Implemented Fleet wide – Industry benchmark information incorporated
  - Strengthened our questioning attitude and lowered entry thresholds
  - Strengthened our screening process, cause analysis tools, and training
- Operability Improvement Project
  - Fleet wide implementation – incorporate industry benchmark information
  - Align IDO and PDO process with industry practice
- Procedure Use & Adherence (PU&A)
  - Engineering Change Manual
- Conservative Decision-Making

Closing Remarks

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