Progress Energy

10 CFR 50.59(d)(2)

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U. S. Nuclear Regulatory Commission
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Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Report of 10 CFR 50.59 Evaluations and Commitment Changes

Ladies and Gentlemen:

In accordance with 10 CFR 50.59(d)(2), Carolina Power & Light Company (CP&L) is providing a report summarizing the 10 CFR 50.59 evaluations of changes, tests, and experiments implemented during the period from August 1, 2010, to July 31, 2012. This report is provided in Enclosure 1. In addition, a summary of commitment changes since August 18, 2010 (i.e., the previous report of 10 CFR 50.59 Evolutions and Commitment Changes), made in accordance with CP&L's commitment management program (i.e., REG-NGGC-0110, "Regulatory Commitments"), is provided in Enclosure 2.

No regulatory commitments are contained in this submittal. Please refer any questions regarding this submittal to Mr. Lee Grzeck, Acting Supervisor - Licensing/Regulatory Programs, at (910) 457-2487.

Sincerely,

FOR
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TMS/tms

Enclosures:

1. Summary of Changes, Tests, and Experiments
2. Regulatory Commitment Change Summary Report

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Summary of Changes, Tests, and Experiments

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Title: Reactor Building Instrument Air Compressor Removal

Evaluation Identification Number: Action Request 196088

Brief Description:

The scope of this Engineering Change (EC) was to decommission the Reactor Building Standby Air Compressors in Units 1 and 2.

Summary of 10 CFR 50.59 Evaluation:

This modification decommissioned the Unit 1 and Unit 2 Reactor Building Standby Air Compressors and initiated associated changes to plant procedures and the Updated Final Safety Analysis Report (UFSAR). The Reactor Building Standby Air Compressors were no longer relied on as a backup to the Non-Interruptible Instrument Air (RNA) header to supply safety significant equipment. This change did not affect other safety significant equipment and had no affect on plant safety response.

Plant References:

Engineering Change (EC) 59731

UFSAR Change Package 07FSAR-005

Title: AOG System Moisture Element Replacement

Evaluation Identification Number: Action Request 320255

Brief Description:

The scope of this change included the design, fabrication, testing and installation requirements along with the output documents such as installation sketches, procedure changes, and database changes for replacing Augmented Off Gas (AOG) System moisture elements 1/2-AOG-ME-013 and 1/2-AOG-ME-018 with a probe that can be replaced on-line.

Summary of 10 CFR 50.59 Evaluation:

This modification activity connected a new moisture probe assembly to the (AOG) system pressure boundary. This assembly permits the removal of the probe on-line without a breach of the AOG pressure boundary. However, should a design bases accident occur while the actual probe or test probe removal was in progress, the gases trapped in the volume of the probe assembly when the root valve is closed may contain part of the accident source term which will then be released to the room and not be held up by the charcoal treatment system as described in the UFSAR. However, this trapped volume activity is insignificant when compared to the activity remaining in the process and released over the duration of the accident and the difference in consequences will be too insignificant to have any quantifiable effect.

Plant References:

EC 55370

Title: Unit 2 Variable Frequency Drive

Evaluation Identification Number: Action Request 377534

Brief Description:

The scope of this change replaced the existing recirculation pump motor-generators and related controls with Variable Frequency Drives (VFDs).

Summary of 10 CFR 50.59 Evaluation:

This modification improved the reliability of the Reactor Recirculation System. The installed VFDs are provided with appropriate safety features, such as redundant controllers and backup protective devices, to prevent the possibility of their acting outside the design basis of the original motor-generators sets. The design parameters for the equipment installed as part of this activity have been deliberately chosen to closely match those of the equipment it is replacing. The operating characteristics are essentially unchanged. A transient analysis has been performed to demonstrate that implementation of the VFD will not significantly impact fuel performance under accident or transient conditions.

Plant References:

EC 68543

UFSAR Change Package 10FSAR-020

Title: Open Access Door To Moderate Steam Tunnel Temperatures

Evaluation Identification Number: Action Request 412802

Brief Description:

In order to moderate steam tunnel temperatures, access door for the back draft damper (2-VA-TB-BDD-3) to the Unit 2 Main Steam Line (MSL) Tunnel was opened to the extent that the lower temperature air from the Unit 2 Turbine Building Breezeway entered the duct and then the Unit 2 steam tunnel due to the negative pressure within the duct. This lowered temperatures in the MSL Tunnel by approximately 2° to 3°F. Clearing this annunciator enabled operators to be able to respond to a valid high Turbine Building tunnel temperature rather than having it masked by the exceptionally high ambient temperatures.

Summary of 10 CFR 50.59 Evaluation:

It was acceptable to open the access door for the 2-VA-TB-BDD-3 to stabilize steam tunnel temperatures. This clears the high temperature alarm without adversely affecting the design function of the steam leak detection system or the operation of the Reactor Feed Pumps and Steam Jet Air Ejectors.

Plant References:

EC 77676

Title: 1-E11-F007B (Minimum Flow Valve) Temporary Modification

Evaluation Identification Number: Action Request 451582

Brief Description:

This change temporarily revised the normal position of valve 1-E11-F007B, RHR loop B minimum flow valve, from closed to open. Typically, the minimum flow bypass valve automatically opens upon sensing low flow in the common discharge line from both pumps of the associated pump pair; and automatically closes whenever the flow in the common discharge line is above the low flow setting. This change allowed the automatic open function of the valve to be disabled. Consistent with the current licensing bases, procedural actions to reposition the valve after the first 10 minutes of an event, for optimal RHR operation were established

Summary of 10 CFR 50.59 Evaluation:

The effects of maintaining the valve in the open position were evaluated and it was determined that there was not a significant impact on plant operation or post-accident response (e.g., emergency diesel generator load, RHR pump runout, net positive suction head (NPSH), torus heating, increased diversion flow, etc.). Consistent with the current licensing bases, procedural actions to reposition the valve after the first 10 minutes of an event, for optimal RHR operation were established, however, manual valve repositioning was not credited for satisfying any safety functions that were previously accomplished by automatic operation.

Plant References:

EC 80291

Title: Troubleshooting For 1B RHR Minimum Flow Monitoring

Evaluation Identification Number: Action Request 451618

Brief Description:

This activity temporarily installed non-safety related test equipment for monitoring of the high and low pressures during troubleshooting of the 1-E11-F007B RHR minimum flow bypass valve. The installed equipment is for monitoring only and is not relied upon for operability of RHR.

Summary of 10 CFR 50.59 Evaluation:

This activity has appropriate equipment and administrative controls to ensure that there is no adverse impact to safety by performing this activity. The installed equipment is for monitoring only and is not relied upon for operability of RHR. The lines where this equipment will be attached are ½ inch. Failure of the installed test equipment could result in leakage past the RHR B loop discharge piping pressure boundary. However, this leakage would be negligible and the RHR pump flow rates have significant margin to the flow required to mitigate accidents described in the UFSAR.

Plant References:

Work Order 1890001

Title: 0PT-13.1 Change To Permit Use Of Test Equipment

Evaluation Identification Number: Action Request 452195

Brief Description:

This procedure revision allowed the connection of test equipment to Reactor Recirculation Jet Pump flow transmitters to record data to be used for satisfying surveillance data addressed in 0PT-13.1 "Reactor Recirculation Jet Pump Operability."

Summary of 10 CFR 50.59 Evaluation:

This activity had appropriate equipment and administrative controls to ensure that there is a negligible risk of this activity and it does not provide an unresolved safety question.

The connection of test equipment did not create any common failure mode and failure of the equipment would not change any environmental conditions for any safety-related equipment. The rating of the equipment exceeds the expected pressures and the effects of failure of the equipment would be bounded by the previously analyzed events for Brunswick.

Plant References:

0PT-13.1

Title: Disabling 2-OG-PS-104B

Evaluation Identification Number: Action Request 468108

Brief Description:

This activity was a temporary change to lift a wire to inhibit the non-safety related 2-OG-PS-104B input to alarm and trip logic. Although the operation of the pressure switches and associated logic are not directly addressed in the UFSAR, performing its function is a change to the trip logic. Before the logic was single failure tolerant, and while the temporary change is in place if 2-OG-PS-105B fails to operate, the trip on low condenser vacuum will not occur.

Summary of 10 CFR 50.59 Evaluation:

The lifting of the wire resulting in the alarm and turbine trip feature of pressure switch 2-OG-PS-104B remained totally bounded by existing accident and transient analysis. In the event of a failure of 2-OG-PS-105B, Operations had sufficient procedural guidance to manually initiate the trip signal if necessary. The affected logic is non-safety related and the alarm and turbine trip feature of 2-OG-PS-104B is not credited in any accident or transient analysis. The bounding analysis assumes that the Main Steam Isolation Valve closure will terminate the event.

Plant References:

EC 81054

Title: Removal Of Degraded Coating on Service Water (SW) Pump Discharge Pipe Spools And Elbows

Evaluation Identification Number: Action Request 510654

Brief Description:

This activity evaluates and controls the removal of the internal ARC©855 coating from the 2A Nuclear Service Water Pump discharge elbow and upstream spool until such time as a permanent repair is implemented.

Summary of 10 CFR 50.59 Evaluation:

Use of uncoated carbon steel will not result in any unacceptable system reliability or performance. Monitoring will ensure that adequate margin exists such that corrosion will not reduce wall thickness below acceptable levels during a given interval between monitoring.

Plant References:

EC 84365

UFSAR Change Package 12FSAR-003

Title: Special Procedure To Allow Placing The VFD's In Local Control

Evaluation Identification Number: Action Request 529113

Brief Description:

This activity created a new procedure (i.e., 2SP-12-201 "RFCS Wiring Checks With VFDs in Local Mode") that disables the communication links between both Unit 2 Variable Frequency Drives (VFDs) and the Recirculation Flow Control System (RFCS) and places the VFDs in local control. This allows checks for loose wires on RFCS to be performed without the risk of impacting VFD operation.

Summary of 10 CFR 50.59 Evaluation:

This activity created a new procedure that opens breakers to disable communication between the VFDs and RFCS, which disables the automatic runbacks such that wire checks can be performed on RFCS without the risk of impacting VFD operation (RFCS will no longer be able to communicate to the VFDs with the breakers opened). Both VFDs will be controlled locally by licensed Operators. During this condition, automatic runback #1 and #2 will not be available. Runback #1 is not credited in any safety analysis. Runback #2 reduces recirculation pump speed, in the event of a loss of one feed pump, to a point where one feed pump is capable of maintaining vessel level and a reactor scram is avoided. During implementation of 2SP-12-201, reactor power will be maintained less than or equal to 72%, obviating the need for runback #2.

Plant References:

Procedure 2SP-12-201

Regulatory Commitment Change Summary Report

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Title: Rosemount Transmitter Loss of Oil Symptom Training

Originating Document:

Letter from J. Cowan (CP&L) to NRC, "Supplemental Response to NRC Bulletin 90-01, Supplement 1 Requested Action 1.f," dated May 26, 1994.

Original Commitment:

The May 26, 1994, letter stated, in part:

> ...CP&L is including symptoms of oil loss associated with Rosemount transmitters during continuing training to I&C maintenance technicians.

Revised Commitment:

The commitment has been deleted.

Basis:

The Rosemount transmitters in question were either replaced with non-oil filled transmitters or were determined to not be subject to oil loss due to the monitoring criteria established in the NRC Bulletin 90-01 response. As such, the need for periodic training on oil loss symptoms has been eliminated.

Title: Residual Heat Removal (RHR) Heat Exchanger Performance Testing

Originating Document:

Letter from James Scarola (CP&L) to NRC, "Revised License Renewal Commitment List," dated December 6, 2005.

Original Commitment:

The December 6, 2005, letter stated, in part:

> The Open-Cycle Cooling Water System Program will be enhanced to require that: ...(6) performance testing of the RHR and Emergency Diesel Generator Jacket Water heat exchangers will be performed to verify heat transfer capability.

Revised Commitment:

The commitment was revised to eliminate the requirement to complete performance testing of the RHR heat exchangers. Periodic inspections will be used as the method to trend heat exchanger condition.

Basis:

Performance testing of the RHR heat exchangers was attempted during refueling outages in 2007 and 2011 for Unit 2 and 2008 for Unit 1 (tested on a 4 year frequency). It was not attempted in 2012 for Unit 1 due to an unplanned shutdown of the unit prior to the scheduled start of the refueling outage.

This testing has shown that proper test conditions cannot be established to support meaningful test results and trending. In order to obtain adequate data a large heat load held constant for an extended period of time is required. Although, early in an outage, the heat load is adequate to support thermal performance testing, previous test data confirms the heat load is not constant during this time period. Intentionally adding heat to the torus or maintaining this heat load for an extended period of time for the purposes of adequate heat exchanger performance trend data is not conservative in regards to nuclear safety.

Electric Power Research Institute (EPRI) Guideline 7552, "Heat Exchanger Performance Monitoring Guidelines," recommends a program of inspections, cleaning, and thermal performance testing as the most effective performance monitoring techniques for safety-related heat exchangers. This guideline also recognizes sole periodic inspections and cleaning as an alternate method when performance testing is not possible. The revised commitment is also consistent with Generic Letter 89-13, "Service Water System problems Affecting Safety-Related Equipment," dated July 18, 1989, which indicates that regular maintenance of a heat exchanger is acceptable in lieu of testing for degraded performance of the heat exchanger.

Based on the above, elimination of the requirement to complete performance testing of the RHR heat exchangers is acceptable.