

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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U.S. Nuclear Regulatory Commission
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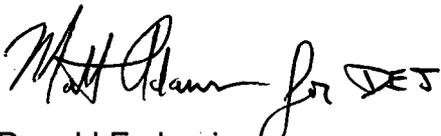
Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
ANNUAL CHANGES, TESTS, AND EXPERIMENTS REPORT
REGULATORY COMMITMENT EVALUATION REPORT

Virginia Electric and Power Company submits the annual report of Changes, Tests, and Experiments pursuant to 10 CFR 50.59(d)(2) and Regulatory Commitment Changes identified in Commitment Evaluation Summaries implemented at Surry Power Station during 2006.

Should you have any questions regarding this report, please do not hesitate to contact Barry Garber at (757) 365-2725.

Very truly yours,



Donald E. Jernigan,
Site Vice President
Surry Power Station

Attachment

Commitments made in this letter: None.

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
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Mr. N. P. Garrett
NRC Senior Resident Inspector
Surry Power Station

IE47

Attachment 1

Surry Units 1 & 2

2006 – 10CFR50.59 Changes, Tests and Experiments

Commitment Evaluation Summary

01/26/06

Description: Generic Letter (GL) 88-17, expeditious actions item number 5, Reactor Coolant System (RCS) Perturbations, required actions to avoid operations that deliberately or knowingly lead to perturbations of the RCS while the RCS is in a reduced inventory condition. In response to GL 88-17 (serial number 88-737), dated January 6, 1989, VEPCO stated that prior to entering reduced inventory, the pressurizer power operated relief valves (PORVs) will be opened to prevent the over pressurization of the RCS. Then the Residual Heat Removal (RHR) suction valves are de-energized after they have been opened to prevent inadvertent closure and subsequent loss of RHR capability. The commitment to venting of the RCS prior to opening and de-energizing the RHR suction valves has been changed to remove the sequencing requirements.

Summary: The intent of the original commitment is maintained by ensuring that, prior to entering a reduced inventory condition, the primary system is vented with the pressurizer PORVs open and the RHR suction valves are de-energized shortly after opening.

06-001 Regulatory Evaluation

02/23/06

Description: Regulatory Evaluation 06-001 reviewed Engineering Transmittal ET-NAF-06-0002, Revision 0, which documents the Peak Clad Temperature (PCT) assessments to the Small Break Loss of Coolant Accident (LOCA) analysis and Large Break LOCA analysis. PCT assessments are associated with the implementation of the Westinghouse Integral Fuel Burnable Absorber (IFBA) fuel product.

Summary: Implementation of the evaluations for both the small break and large break LOCA analysis documented in ET NAF-06-0002 and Calculation SM-1294, revision 0, Addendum I, Surry Large Break LOCA Analysis with 1981 BASH Evaluation Model: Evaluation of Fuel Containing IFBA, did not increase the probability of occurrence of a small or large break LOCA, or create the possibility of an accident of a different type than previously evaluated in the UFSAR. The activity does not, by itself, make any change to allowable Surry Unit 1 and 2 operating conditions defined by Technical Specifications and operating procedures or to any plant design features. The evaluations are in accordance with NRC-approved methodologies that are listed in the Core Operating Limits Report (COLR), and do not constitute a change in method of evaluation.

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06-002 Regulatory Evaluation

03/30/06

Description: Regulatory Evaluation 06-002 reviewed Engineering Transmittal ET-NAF-06-0001, Revision 0, which documents the revised nuclear safety analysis related to a postulated Control Rod Ejection Accident at Surry including the use of Integral Fuel Burnable Absorber (IFBA) fuel assemblies. The reanalysis of the Surry Rod Ejection event has been performed to cover both IFBA and non-IFBA 15x15 Surry Improved Fuel assemblies.

Summary: The reanalysis accommodates higher values of ejected rod worth and post ejection F_Q peaking factors, which are expected to result from the use of reload core patterns using IFBA fuel. The revised analysis demonstrates that operation within the limits prescribed by the Technical Specifications ensure that all generic Condition IV and Regulatory Guide 1.77 acceptance criteria, as well as Topical Report VEP-NFE-2-A, VEPCO Evaluation of Control Rod Ejection Transient, acceptance criteria are met with respect to a postulated Control Rod Ejection accident.

06-003 Regulatory Evaluation

04/20/06

Description: Regulatory Evaluation 06-003 reviewed the performance of maintenance on the Unit 1 or Unit 2 "A" Main Station Batteries during empty vessel or when the water depth in the reactor cavity is equal to or greater than 23 feet above the top of the reactor pressure vessel flange. Maintenance on "A" Main Station Batteries was previously performed only during empty vessel.

Summary: The issue considered for this maintenance activity was a loss of offsite power (LOOP) occurring while the "A" Battery is disconnected from the "A" DC Bus. Procedures were revised to provide a controlled means to restore the affected DC Bus and the associated Emergency Bus. Continuous source range monitoring is maintained during these activities. Decay heat removal restoration and continuous source range monitoring continues to maintain compliance with Technical Specifications and with the cavity flooded at equal to or greater than 23 feet, sufficient time exists to recover AC power.

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06-004 Regulatory Evaluation 04/25/06

Description: Regulatory Evaluation 06-004 reviewed the freeze sealing of the pipe down stream of an accumulator check valve (1-SI-147) in the Reactor Coolant System to allow maintenance on the valve.

Summary: The freeze seal is performed when the core is off loaded, all fuel is in the Spent Fuel Pool, and the Fuel Transfer Valve is closed, therefore, no change in the frequency or severity of an Accident or Malfunction is created.

Commitment Evaluation Summary 04/25/06

Description: Generic Letter (GL) 91-11 required licensees to impose time limits on vital bus inverter maintenance. In response to GL 91-11 (serial number 91-456), dated January 31, 1992, VEPCO committed to limit completion time for vital bus inverter maintenance. A previous evaluation in 2004 extended the completion time to 7-days and imposed restrictions on Switchyard maintenance during inverter maintenance. This commitment evaluation eliminates restrictions on Switchyard maintenance during vital bus inverter maintenance.

Summary: A risk evaluation supported the removal of the Switchyard maintenance restriction. Restrictions continue to be required on planned maintenance or surveillance on emergency diesel generators and on RPS/ESFAS systems.

06-005 Regulatory Evaluation 05/10/06

Description: Regulatory Evaluation 06-005 evaluated the capping and removal from service one flux thimble tube (1-RC-TW-J3).

Summary: The flux thimble tube is capped with the same material and construction as the fitting to ensure that the RCS pressure boundary is maintained. In addition, incore instrumentation is maintained sufficient to meet Technical Specification requirements.

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06-007 Regulatory Evaluation

05/17/06

Description: Regulatory Evaluation 06-007 reviewed the completion of 1H Bus logic testing, with various components unavailable, through use of procedurally controlled Temporary Modifications (TM). This activity modifies control circuit interlocks for the 1C charging pump normal and alternate feeder breaker to defeat the interlocks that prevent both breakers from being racked to test or connect simultaneously. This ensures the 1C charging pump remains available for operation during the time the H breaker is being tested.

Summary: With the unit shutdown, charging pumps are not required to be operable; however, administrative procedures require one charging pump to be available. The procedurally controlled TM ensures that the 1C charging pump remains available for operation if required during the time that the H6 breaker is being tested. The TM is removed as directed by procedure prior to the unit exceeding 200°F.

06-008 Regulatory Evaluation

08/24/06

Description: Regulatory Evaluation 06-008 reviewed the replacement of Circulating Water (CW) 96" inlet motor operated valves (MOVs). The CW inlet MOVs isolate to maintain canal level inventory or to mitigate flooding in the turbine building. The new MOV valves are manufactured from 316L stainless steel. Surry has experienced several through-wall corrosion failures of fabricated 316L stainless steel materials in the service water system that was determined to be microbiologically induced corrosion (MIC). MIC attack of these valves can increase the potential for the valve to fail in the open position or experience excessive through wall leakage. The new valves have been treated to increase corrosion resistance; however, the materials have not been tested to determine the effectiveness of the resistance in full flow and in long term stagnant conditions.

Summary: The change is considered acceptable because valve damage due to MIC is typically identified prior to internal valve failure, excessive external leakage, or structural valve body damage. In addition, Engineering is providing ongoing valve inspection and confirmatory passivated 316L weld coupon testing per Engineering Transmittal MAT-06-0001. The confirmatory testing includes a temporary installation of passivated 316L weld coupons immersed in river water located at the Surry Low Level.

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06-006	Regulatory Evaluation	05/12/06
06-009	Regulatory Evaluation	10/26/06

Description: Regulatory Evaluations 06-006 and 06-009 reviewed a design change to disable the auto-open function for the Auxiliary Feedwater (AFW) MOVs that feed the “B” steam generators (SGs) on Surry Units 1 & 2. A design issue was discovered where the AFW pumps, if called upon to perform their function, could reach unanalyzed high flow conditions when the RCS is between 350°F and 500°F. Restricting AFW flow by isolating “B” SG resolved the potential high flow conditions.

Summary: The “B” SG MOVs are closed and the auto-open function disabled between RCS temperatures of 350°F and 500°F. When the RCS temperature is above 535°F, all six SG MOVs are maintained full open with the 4 MOVs to A and C SG retaining the auto-open feature. The above configurations continue to comply with Technical Specifications which requires a minimum of two operable SGs in non-isolated loops when the average RCS temperature is above 350°F and three RCS loops to be in service at power operations. The proposed configuration ensures that the AFW system can perform its design function for all design basis accidents with consideration of a single failure.

06-010	Regulatory Evaluation	11/02/06
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Description: Regulatory Evaluation 06-010 reviewed a design change to add a permissive signal to disable Overspeed Protection Control (OPC) circuit on the Electro-Hydraulic Control (EHC) System when one of two generator output breakers is closed and the main Generator output is 5% or greater. This modification prevents closure of the Governor and Intercept Valves by an OPC actuation when the main turbine is at grid frequency.

Summary: A spurious OPC actuation was identified as a single point vulnerability (SPV). The addition of a permissive signal to the OPC circuit such that the circuit is disabled when the main generator is synchronized on the grid has no impact on the evaluation because this protection is not needed when the generator is synchronized with the grid. The turbine can not overspeed when the generator is synchronized with the grid. The OPC circuitry is enabled when the main generator is not supplying electricity to the grid.

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06-011 Regulatory Evaluation

10/17/06

Description: Regulatory Evaluation 06-011 reviewed the design changes necessary to comply with the NRC Generic Safety Issue (GSI) 191 and Generic Letter (GL) 2004-02. The existing containment sump screens are to be replaced with a new strainer to ensure that post-accident debris blockage will not impede the operation of the Low Head Safety Injection (LHSI) and the Recirculation Spray (RS) systems in the recirculation mode during LOCA or other HELB accidents.

Summary: Surry's existing containment sump screen has been evaluated and determined to be unsatisfactory with respect to the GSI 191 and GL 2004-02. The existing containment sump screens are being replaced with a new strainer to address the GSI-191 concerns. The requirements provided in the new methodology to determine the new strainer design basis are more stringent than previously provided by the NRC. The resulting design is a more conservative design than the existing screen design.