VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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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. Attachment 1 provides a description and summary of the Regulatory Evaluations completed in 2016. There were no Regulatory Commitment Change Evaluations completed in 2016.

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

Very truly yours,

Douglas C. Lawrence,
Director Nuclear Safety & Licensing
Surry Power Station

Attachment

Commitments made in this letter: None

cc: United States Nuclear Regulatory Commission, Region II
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NRC Senior Resident Inspector
Surry Power Station

IE47
NRR

Attachment 1

Surry Units 1 & 2

10 CFR 50.59 Changes, Tests, and Experiments

16-001 **Regulatory Evaluation** 01/21/2016

Description:
Regulatory Evaluation 16-001 reviewed the implementation of the statistical / deterministic F∆H (Enthalpy Rise Hot Channel Factor) limits of 1.635/1.70 for Surry Unit 2, which were increased from the previous F∆H limits of 1.56/1.62. Included in the review were the implementation of several Departure from Nucleate Boiling (DNB) accident analyses and the reactor protection verification that were reanalyzed to support the higher F∆H limits.

Summary:
In order to improve station flexibility, the increased statistical / deterministic F∆H limits of 1.635/1.70 for Surry Unit 2 were implemented during fuel cycle 27. The DNB accident analyses and verification of the reactor protection setpoints were reanalyzed to support the increased F∆H limits. In all cases, the revised DNB analyses were confirmed to meet all applicable Critical Heat Flux (CHF) correlation limits listed in the Surry Updated Final Safety Analysis Report (UFSAR) and Surry Technical Specifications (TS). The increased F∆H limits and revised accident analyses have been shown to not impact the associated design function. Design Basis Limits for Fuel Product Barriers (DBLFPB) are not exceeded or altered, and this activity does not result in a departure in a method of evaluation as described in the Surry UFSAR.

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SPS0-EVAL-2016-0001 **Regulatory Evaluation** 04/18/2016

Description:
Regulatory Evaluation SPS0-EVAL-2016-001 was performed for the activity of replacing the Westinghouse Reactor Coolant Pump (RCP) seals with Flowserve N9000 (N-Seal) RCP seals on all six Surry RCPs. The new seal design was chosen in order to improve reliability, and also to reduce susceptibility to and consequences of failure under adverse conditions.

Summary:
RCP seals provide a portion of the Reactor Coolant System (RCS) pressure boundary between the rotating RCP shaft and the stationary RCP flange. As such, RCP seal failure could constitute a Loss of Coolant Accident (LOCA). Therefore, the RCP seal design is critical under conditions of loss of seal injection or component cooling water flow which could occur during station blackout and certain fire conditions. The NRC has expressed confidence in the quality of the N-Seal (which has a history of good performance at Combustion Engineering plants), and recommended that other stations consider potential failure modes as part of the conversion process to the N-Seal. Surry has reviewed the potential for additional failure modes and concluded that no other modes than LOCA are credible.

Technical reports which model N-Seal failure have demonstrated that failure probability of the N9000 seal is significantly lower than the Westinghouse seal they replaced. Additionally, if a malfunction were to occur under worst case scenarios, the leakage of the N9000 seal would be considerably less than that of the Westinghouse seal.

The Flowserve seals are of a similar design and perform fundamentally the same function as the Westinghouse seals. As such, the new seals do not create the possibility of a component malfunction with a result different that previously evaluated. Since under accident conditions leakage from the RCS would be considerably less for the N9000 seal compared with the Westinghouse seal, it is concluded that Radiological Consequences of a Loss of Coolant Accident (from seal failure) is not adversely affected by use of the N9000 seals. Likewise, all pressure boundary components of the N9000 seal satisfy existing design criteria, thereby not altering any fission product barrier limit.

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SPS0-EVAL-2016-0002 **Regulatory Evaluation** 02/09/2016

Description:
The Surry Reserve Station Service Transformers (RSSTs) are reaching the end of their design service lives and will be replaced. This Regulatory Evaluation reviewed the replacement of transformers and associated support equipment. Specifically, this evaluation reviews the microprocessor based RSST On-Load Tap Changers (OLTC) and digital relays for pilot wire differential fault protection, which will replace the existing analog devices.

Summary:
A software or firmware failure associated with the new RSSTs could adversely affect the 4KV station emergency electrical bus primary off-site power supply via the RSSTs due to potential common mode failures. Failure of these components could result in loss of off-site power or incorrect voltage supply to the associated emergency buses. Although the possible failure causes are potentially different (because the equipment design is different, e.g. digital vs. analog), analyzed failure modes (loss of the RSST and resultant loss of off-site power) are unchanged. Therefore, the consequences of analyzed accidents or malfunctions are unchanged. Also, since failure modes of the new RSSTs are within the bounds of previous analyses, no new accident initiator or accident of a different type is created.

The new transformer equipment has been designed, analyzed, and tested to support this specific application at Surry, and it is expected that reliability of the RSSTs and protective relaying will be enhanced. Therefore, the likelihood of occurrence or consequences of a malfunction of structures, systems, and components (SSC) important to safety is not more than minimally increased.

Surry RSSTs are not credited for accident mitigation, therefore no fission product barrier limits will be altered.

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SPS0-EVAL-2016-0003 **Regulatory Evaluation** 05/05/2016

Description:
Surry UFSAR was changed in order to clarify the proper fuel clad stress evaluation methodology for Optimized ZIRLO fuel cladding. The UFSAR previously referenced only the clad stress methodology in Addendum 1 to WCAP-10125-P-A, which did not accurately represent compressive stress limits for Optimized ZIRLO cladding under all conditions. A reference was therefore identified for standard Westinghouse clad stress methodology (WCAP-12610-P-A), and it was clarified that only this methodology is being applied to fuel with Optimized ZIRLO cladding. A 10 CFR 50.59 evaluation was required due to a change in the methodology for determining fuel cladding stress design basis limits described in the Safety Analysis Report.

Summary:
As part of the 2009 fuel design upgrade at Surry, the UFSAR was modified to add reference to NRC-approved clad stress criterion for Westinghouse fuel (Addendum 1 to WCAP-10125-P-A). That UFSAR change indicated that the revised (ASME-based) fuel clad stress criterion would be suitable for analyzing all cladding types used in both Surry units, including Optimized ZIRLO cladding.

In 2015, Westinghouse notified Dominion that, due to the lower unirradiated yield strength of Optimized ZIRLO cladding, the compressive stress limits at beginning of core life are not met for Optimized ZIRLO cladding using the clad stress methodology described in Addendum 1 to WCAP-10125-P-A. However, this methodology is conservative, and certain compressive conditions present in the fuel rod design are not credited in the clad stress evaluation using this methodology.

For fuel with Optimized ZIRLO cladding at Surry, Westinghouse is now using the conservative clad stress methodology discussed in WCAP-12610-P-A. Since this methodology has also been approved by the NRC, this change did not constitute a departure from the methods used to establish the design basis for or used in the safety analysis of the fuel rod cladding. Likewise, the cladding methodology reference change did not result in a design basis limit for a fission product barrier being exceeded or altered.

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SPS1-EVAL-2016-0005 **Regulatory Evaluation** 04/14/2016

Description:
Evaluation SPS1-EVAL-2016-0005 reviewed the implementation of the accident analyses and reactor protection verification that were reanalyzed to support the higher statistical / deterministic FΔH (Enthalpy Rise Hot Channel Factors) limits of 1.635/1.70 for Surry Unit 1 (an increase from the previous FΔH limits of 1.56/1.62).

Summary:
In order to improve station flexibility, the increased statistical / deterministic FΔH limits of 1.635/1.70 were reviewed for Unit 1 before fuel cycle 28. In order to support the increased FΔH limits, the Departure from Nucleate Boiling (DNB) accident analyses and verification of the reactor protection setpoints were reanalyzed. In all cases, the revised DNB analyses were confirmed to meet all applicable CHF correlation limits listed in Surry UFSAR and Surry TS. The increased FΔH limits and revised accident analyses have been shown to not impact the associated design function. Design Basis Limits for Fuel Product Barriers (DBLFPB) are not exceeded or altered and this activity does not result in a departure in a method of evaluation as described in the Surry UFSAR.

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Surry Units 1 & 2

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SPS0-EVAL-2016-0007 **Regulatory Evaluation** 08/04/2016

Description:
Evaluation SPS0-EVAL-2016-0007 reviews installation and operation of voltage unbalance relays on the potential transformers (PTs) for 4kV emergency buses in order to detect an Open Phase Condition (OPC) on those buses. Activation of the OPC protection scheme will result in an alarm in the Main Control Room and disconnection of the affected 4kV emergency bus from the degraded offsite power source. The emergency diesel generator (EDG) will start and load on the affected emergency bus, supplying power to run necessary safety equipment.

Summary:
The addition of protective relays to the Surry 4kV emergency buses could potentially affect a previously analyzed design basis Structures, Systems, and Components (SSC) function, since installation of the OPC system increases the probability of emergency bus isolation and EDG start in the event of a spurious actuation of the OPC system. Since the actuation of the OPC relays results in automatic starting of the associated EDG, load shedding, and re-loading of the emergency buses, the relay actuation also results in increased challenges to systems and components important to safety. However, if actuation is in response to a valid OPC signal from the new protection system, these automatic actions will reduce the likelihood of a malfunction of systems and components important to safety, because the 4kV emergency buses and equipment powered from them are then protected from the damaged equipment that caused the open phase condition. A spurious actuation of the OPC protection system is improbable, and therefore, the likelihood of unnecessarily challenging systems and components important to safety is not more than minimally increased. Also, spurious actuation of the OPC system does not have the potential to contribute to the initiation of accidents previously evaluated in the UFSAR.

Methods of analysis and Design Basis Accident analysis results are unchanged by this modification. Also, the effect of the proposed activity does not create the possibility for an accident or malfunction of a different type than previously evaluated in the license basis. The UFSAR-described dose analyses remain bounding and valid.

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SPS1-EVAL-2016-0009 **Regulatory Evaluation** 10/06/2016

Description:
Evaluation SPS1-EVAL-2016-0009 reviewed the loading and operation of eight AREVA AGORA-5A-I Lead Test Assemblies (LTAs), the application of existing NRC-approved methods and analyses of record beyond the bounds of the original NRC safety evaluations in order to evaluate the LTAs, and the use of AREVA methods to assess LTA fuel rod design and assembly performance against AREVA NRC-approved design criteria.

Summary:
This evaluation was deemed necessary in the 10 CFR 50.59 Screen because the AGORA LTAs are a test/experiment not specifically described in the Surry Updated Final Safety Analysis Report (UFSAR) and outside the reference bounds described in the UFSAR, and are evaluated with methods outside the bounds of the original NRC safety evaluations or not previously implemented at Surry. Cycle-specific and generic engineering evaluations concluded that the AGORA LTAs do not increase the frequency of occurrence of an accident or an SSC malfunction, and have been shown to meet all design limits so that the consequences of an accident or SSC malfunction are not increased. No new accident or SSC malfunctions are created by the AGORA LTAs, due to the similarity with the resident fuel product. Engineering evaluations determined that no DBLFPBs were exceeded or altered. There was no departure from a method of evaluation described in the UFSAR, as appropriate restrictions have been imposed (limited number of LTAs in non-limiting core locations in accordance with Surry Technical Specification).