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**

Dear Sir or Madam,

Virginia Electric and Power Company hereby submits the annual report of Changes, Tests, and Experiments pursuant to 10CFR50.59(d)(2) implemented at Surry Power Station. Attachment 1 provides the descriptions and summaries of Regulatory Evaluations and Regulatory Commitment Change Evaluations completed in 2018.

Should you have any questions regarding this report, please contact Barry Garber at (757) 365-2725.

Very truly yours,

Robert M. Garver II
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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245 Peachtree Center Avenue, NE
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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
Regulatory Commitment Evaluations

**SPS0-EVAL-2017-0004 Regulatory Evaluation 02/15/2018**

Description:
This evaluation reviewed the following two items: 1) adoption of Technical Specification Task Force (TSTF) Traveler TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Technical Specification" and 2) update of the Alternative Source Term (AST) analyses bases. The proposed Technical Specification (TS) changes replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on dose equivalent XE-133 and would only take into account the noble gas activity in the primary coolant. The TS are also being revised to change the units for the specific activity of the secondary coolant system, as well as the required actions if the activity is exceeded. The AST analyses bases are updated for new codes, revised atmospheric dispersion factors (X/Qs), new fuel handling accident fuel rod gap fractions, control room isolation operator action time, and elimination of the locked rotor accident (LRA) dose consequences.

Summary:
The evaluation determined the following:

1. Changes to the TS require NRC review and approval pursuant to 10 CFR 50.92. As such, the proposed changes were submitted to the NRC for review and approval by letter serial number 18-069, dated March 2, 2018.

2. NRC review and approval of the updated AST analyses bases is required pursuant to the requirements of 10 CFR 50.59, which specifies that a departure from a method described in the UFSAR, such as the design basis radiological consequence analyses, shall be submitted for approval unless the changes to the elements of the method meet certain requirements. The proposed changes for radiological events are replacing the computer code used to calculate dose, revising X/Qs for Control Room and offsite receptors (including the computer code and method used to determine Control Room X/Qs for SG releases), replacing the computer code used to determine core inventory, changing FHA gap fraction methodology, and removing the LRA from the radiological design basis.

   The cumulative effect of input parameter changes for the LOCA, FHA, SGTR, and the MSLB are deemed to constitute more than a minimal increase in consequences and therefore require NRC review and approval prior to implementation. This includes deletion of the dose consequences of the LRA from the radiological design basis because fuel damage is not predicted, in accordance with RG 1.183 Appendix G.2. In general, these changes in methodology result in increases in margin to the dose consequences limits and as a result are adverse and require NRC review and approval prior to implementation. The changes in computer code for the dose calculation code from LOCADOSE to RADTRAD-NAI and the change in source term code from ORIGEN2 to ORIGEN-ARP are exceptions. These changes resulted in dose consequences that are essentially the same or are conservative, but were submitted for NRC review for convenience. The NRC was not asked to review changes to the source term released for the VCT rupture analysis. The change in VCT release source term was primarily the result of using the ORI GEN-ARP based core inventory that the NRC is being asked to approve. The ORIGEN-ARP core inventory was used to derive a 1% failed fuel coolant activity that was used to derive the RCS TS coolant activities for the SGTR and MSLB analyses. The change to the VCT rupture analysis results in dose consequences that remain less than 0.5 Rem whole body and are essentially the same as the current design basis analysis. These proposed changes were analyzed and resulted in acceptable consequences, meeting the criteria as specified in 10 CFR 50.67 and RG 1.183, but did not meet the requirements for implementation under 10 CFR 50.59.

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**SPS-EVAL-2014-003, Revision 1** **Regulatory Evaluation** **04/03/2018**

Description:
Regulatory Evaluation 14-003, Revision 1, reviewed use of a temporary Service Water (SW) flow path from the discharge of the Component Cooling Heat Exchangers (CCHX) so that maintenance on certain SW equipment could be performed during the 2018 Surry Unit 1 refueling outage. Unit 2 was operating at power during the time the CCHX SW temporary line was in service. The modification was required to maintain adequate component cooling (CC) to remove residual and sensible heat from the operating and shutdown units and from the spent fuel pool. The CC and SW design functions and basic configurations were not altered as a result of using the temporary discharge line.

Summary:
The temporary CCHX SW discharge line was designed with the attributes of the normal CCHX SW discharge line, except for complete missile and heavy load drop protection. For this reason, the possibility of flooding or loss of heat removal due to missile or heavy load damage was controlled by the implementation of compensatory measures and station procedures. Continuous administrative controls were in place to monitor for flooding and to isolate SW flow as necessary. As under normal conditions, automatic actuations and procedures were in place to isolate SW supply to the CCHXs in the event of flooding. Additionally, the CCHXs are not credited in accident mitigation. For these reasons, the frequency or likelihood of occurrence of an accident or malfunction of a system, structure, or component (SSC) important to safety was not more than minimal. Also, the consequences of an accident or malfunction of a SSC important to safety were not more than minimally increased. A malfunction of the temporary CCHX SW discharge line would have the same result as a malfunction of the normally in service CCHX SW discharge line. Therefore the probability of malfunction with a different result than previously evaluated was not created.

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**SPS1-EVAL-2018-0001** **Regulatory Evaluation** **05/03/2018**

Description:
As part of a 2018 Unit 1 refueling outage maintenance testing activity associated with the Open Phase Condition Safety Related Protection and Detection System, a motor operated valve for one train of service water (SW) supply to component cooling heat exchangers (CCHX) was deenergized in the open position. Under normal operating conditions, the valve receives a closed signal in the event of turbine building flooding or upon loss of offsite power (which could result in low intake canal inventory). In order to provide a strategy for coping with these events, manual actions to isolate SW to the CCHXs were proceduralized for implementation during the time power was removed from the SW supply motor operated valve. A regulatory evaluation was performed since manual actions to close qualified isolation valves would substitute for automatic actuation of a motor operated valve in the event of low intake canal level.

Summary:
Under conditions of low intake canal level, a time limit of one hour for isolation of component cooling SW loads had been previously analyzed as acceptable. As such, a one hour time constraint is included in the abnormal procedure for loss of intake canal level. A one hour completion time for manual actions associated with this activity was proceduralized as well, consistent with the analyzed one hour limitation. Since the analysis assumes that a motor operated SW supply valve may fail to close, the likelihood of occurrence of a malfunction of a SSC important to safety was not more than minimally increased by this activity.

Outage risk plans and work controls ensured that activities associated with Open Phase Condition Safety Related Protection and Detection System modifications would not initiate a malfunction of a different type or the frequency of occurrence of an accident previously evaluated, including turbine building flooding.

Since proceduralized controls to isolate SW to the CCHXs were established within the assumptions of Surry design bases, this activity would not result in more than a minimal increase in the consequences of an accident or malfunction of equipment. Likewise, this activity would not result in exceeding or altering a design basis limit for a fission product barrier.

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**SPS0-EVAL-2016-0002, Revision 1 Regulatory Evaluation 09/06/2018**

Description:
Three Reserve Station Service Transformers (RSST) are the primary sources of offsite power to the Unit 1 and Unit 2 station emergency busses. The RSSTs are original plant equipment and are near the end of their dependable service lives. This evaluation reviews the replacement of existing RSSTs and associated equipment, and is necessary since microprocessor based RSST On-Load Tap Changers (OLTC) and digital relays for pilot wire differential fault protection will be installed in place of existing analog equipment. These changes could introduce the potential for software or firmware common mode failure which could impact the power supply to the emergency busses from RSSTs. Additionally, the RSSTs will be placed on temporary control power for a short duration while existing control power feeders are reconfigured. Failure of this temporary control power supply would be no different from failure of the current control power supply.

Summary:
Potential failure modes of the digital OLTCs and digital relays could result in loss of offsite power or incorrect voltages to the associated emergency busses. The results of these failure modes are unchanged by this modification. The SPS accident analyses are bounding for the loss of offsite power (including loss of RSSTs) for all analyzed accident conditions. As such, the failure of an RSST to supply power to mitigate accident consequences is not credited. Similarly, where a worst case accident condition assumes that offsite power remains available, the consequences of the accident analyses remain bounding. Although digital equipment failure causes are potentially different, the failure modes are no different from the presently installed analog equipment. Therefore, the consequences of an accident or malfunction of an SSC important to safety are not changed. Likewise, since no failure modes are changed, no new accident initiators are created, and the possibility of an accident of a different type will not be created.

The new equipment has been fully tested, analyzed, and designed to support the RSST application at Surry Power Station. The equipment is proven reliable and environmentally suited to ambient conditions. Therefore, the frequency of occurrence of an accident or incident is not more than minimally increased. Also, the activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SCC important to safety.

Since RSSTs are not credited for accident mitigation, no fission product barrier limits will be exceeded or altered.

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**Commitment Evaluation Summary** **08/28/2018**

Original Commitment Summary:
In response to NRC bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," Surry committed (in letter Serial Number 96-206 dated 5/13/1996) to initiate appropriate licensing action and obtain NRC approval if it is necessary to perform activities involving the handling of heavy loads over spent fuel, over fuel in the reactor core, or over safety-related equipment.

Revised Commitment Summary:
Movement of heavy loads as part of a maintenance activity will be managed under 10CFR50.65(a)(4), and those that are part of a design change will be evaluated in accordance with 10CFR50.59 to determine if prior NRC approval is required.

Justification:
NEI 96-07 Revision 1, Update 4 (April 2001) states, "Movement of heavy loads is typically part of a maintenance activity that, going forward, will be assessed and managed under 10CFR50.65(a)(4). Together with 10CFR50.59(c)(4), which provides that if more specific requirements apply to control of an activity, 10 CFR 50.59 need not also be applied; these new requirements supersede the conclusion of NRC bulletin 96-02 that such activities constitute "unreviewed safety questions" under 10CFR50.59 and therefore a license amendment must be submitted."

However, if the heavy load movement is controlled by design change instead of a maintenance activity, the design change will include a 10 CFR 50.59 review that will determine if prior NRC approval is required.