VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 15, 2019

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 19-131
NRA/GDM: R0
Docket Nos.: 50-280/281
License Nos.: DPR-32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST
ADDITION OF 24-HOUR COMPLETION TIME FOR
AN INOPERABLE REACTOR TRIP BREAKER

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion Energy Virginia) requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station (Surry) Units 1 and 2. The proposed change revises Action 8.A associated with Item 18 in Surry TS Table 3.7-1, "Instrumentation Systems," to provide a completion time (CT) of 24 hours to restore an inoperable Reactor Trip Breaker (RTB) to operable status. The CT is an addition to the current Action 8.A requirement to be in Hot Shutdown within 6 hours if one RTB is inoperable. The 24-hour CT provides additional time to perform maintenance activities at power while minimizing the risk associated with the loss of the component function.

Attachment 1 provides a discussion and assessment of the proposed change, including the results and conclusions from the supporting Probabilistic Risk Assessment (PRA). As discussed in Attachment 1, external hazards have been screened as not significant risk contributors for this LAR evaluation. Dominion Energy Virginia expects to submit a more detailed assessment of external hazards later this year as part of a LAR to implement 10 CFR 50.69 at Surry Power Station; however, we do not anticipate any impact to the conclusions herein that external events are not significant risk contributors. Marked-up and proposed TS pages reflecting the proposed change are provided in Attachments 2 and 3, respectively. Attachment 4 provides a discussion of the technical adequacy of the PRA model that was used to support this LAR.

We have evaluated the proposed amendment request and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is included in Attachment 1. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released off-site or any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The proposed TS change included in this LAR has been reviewed and

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approved by the Facility Safety Review Committee.

Dominion Energy Virginia requests approval of the proposed TS change by May 31, 2020, with a 60-day implementation period.

Should you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Respectfully,

Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

Commitments contained in this letter:

1. In order to maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one Reactor Trip Breaker is inoperable.

Attachments:

1. Discussion of Change
2. Marked-up Technical Specifications Page
3. Proposed Technical Specifications Page
4. Probabilistic Risk Assessment Quality Evaluation

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mr. Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 15 day of May, 2019.

My Commission Expires: March 31, 2022

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MY COMMISSION EXPIRES MARCH 31, 2022

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Surry Power Station

Attachment 1

DISCUSSION OF CHANGE

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

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DISCUSSION OF CHANGE

1.0 SUMMARY DESCRIPTION

The proposed change revises Action 8.A associated with Item 18 in Surry Power Station (SPS) Technical Specifications (TS) Table 3.7-1, "Reactor Trip Instrument Operating Conditions," for one inoperable Reactor Trip Breaker (RTB). The current Action requires the plant to be in at least Hot Shutdown within 6 hours if one RTB is inoperable. The revised Action provides a completion time (CT) of 24 hours to restore an RTB to operable status in addition to the 6-hour Hot Shutdown requirement. This change is consistent with the established CT of 24 hours for the automatic trip logics. Implementation of the 24-hour CT provides time to perform maintenance activities on a single RTB during power operation. As such, the additional 24 hours will reduce unnecessary plant shutdowns and subsequent start-ups associated with TS compliance. The CT will also help avoid requests for discretionary enforcement to remain at power when the time to complete a RTB repair or restoration activity could exceed 6 hours.

The 24-hour CT to restore a single RTB to operable status was approved by the NRC for Westinghouse plants in Westinghouse topical report, WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," Revision 1 (hereafter referred to as WCAP-15376). Based on NRC acceptance as stated in TSTF-411-A, Rev. 1, "Surveillance Test Interval Extensions for Components of the Reactor Protection System," (hereafter referred to as TSTF-411), new test intervals and CTs were incorporated into NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4. This license amendment request (LAR) uses WCAP-15376 and the Surry-specific probabilistic risk assessment (PRA) as justification for the proposed CT addition and is consistent with NRC approved TSTF-411.

The supporting probabilistic risk assessment (PRA) performed for the 24-hour CT concluded that the increase in risk associated with the proposed change is consistent with acceptance guidelines for a permanent CT change contained in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Bases," dated July 1998, and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998. The supporting PRA demonstrates that defense-in-depth will not be significantly impacted by allowing a single RTB to be inoperable for up to 24 hours. Although a detailed review of PRA importance metrics from the Tier 1 PRA model did not reveal any risk significant maintenance configurations, to maintain appropriate measures of defense in depth, no maintenance will be planned on the Anticipated Transient without Scram (ATWS) Mitigating System Actuation Circuitry (AMSAC) system while one RTB is inoperable. [Regulatory

Commitment] No additional enhancements, procedure revisions or compensatory actions are recommended from the Tier 2 evaluation.

2.0 DETAILED DESCRIPTION

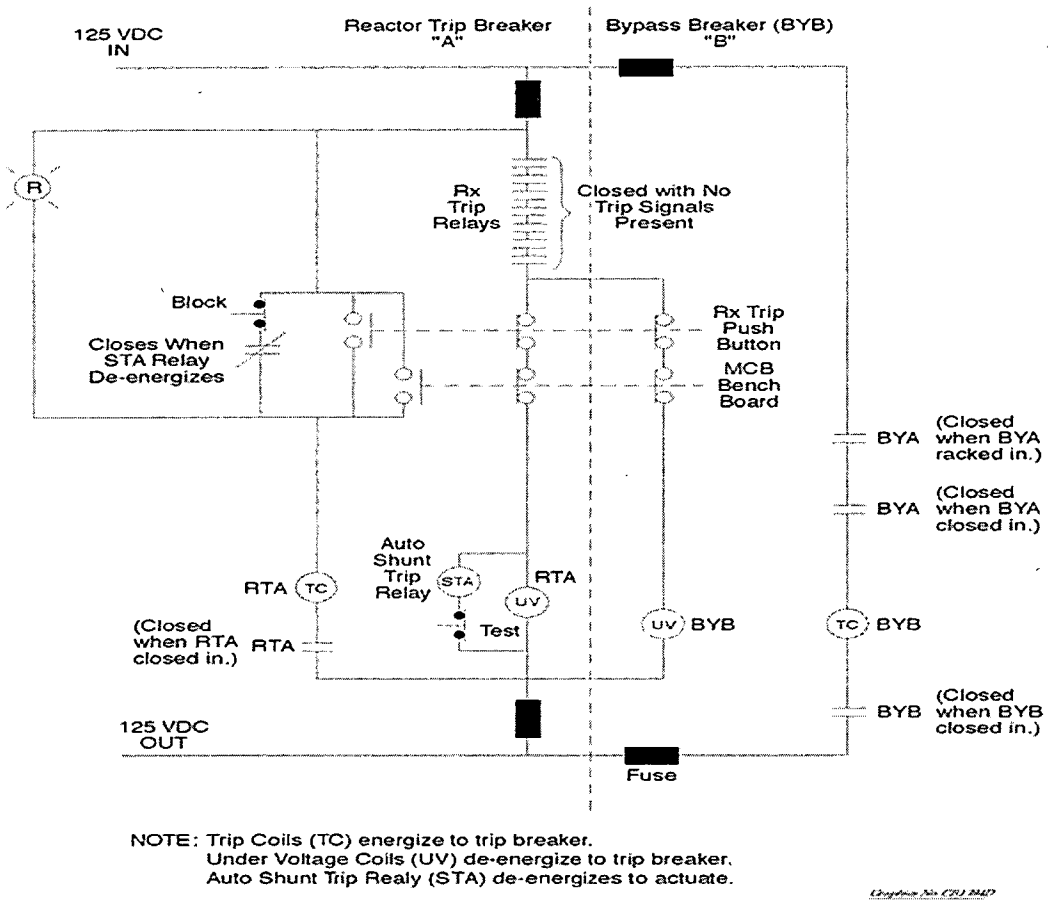
2.1 System Design and Operation

The Reactor Protection System (RPS) is designed to guarantee the integrity of the reactor systems during normal operating and accident conditions. The RPS supplies signals to automatically trip the reactor whenever plant conditions approach safety limits that could challenge the Reactor Coolant System or fuel integrity. There are two independent trains of logic circuitry and RTBs in the RPS. Each redundant train receives separate logic signal inputs and is energized from separate AC power sources. When a trip demand is sensed by either logic train, a trip signal causes the associated RTB to open. The two trains of RTBs operate in series such that opening either train's breaker interrupts power to all control rod drive mechanisms (CRDMs), causing them to release all control rod assemblies, which fall by gravity into the core. The SPS RPS employs a relay protection system versus a solid state protection system.

Each breaker is opened by an undervoltage trip coil, which senses the loss of power when a reactor trip demand is generated. A shunt trip relay operates in parallel with the undervoltage coil and is installed as a redundant means to automatically trip the RTB upon receipt of a trip signal from the RPS. Upon receipt of a trip signal, power is interrupted to the shunt trip relay (shown as STA in the figure below), and contacts from the STA close to energize the RTB shunt trip attachment, which trips open the RTB. Either the undervoltage coil or the shunt trip attachment is sufficient by itself to open the RTB, thus providing diverse reactor trip mechanisms.

Each RTB is equipped with a Reactor Trip Bypass Breaker (BY) to allow maintenance and testing of the RTB while the unit is at power. Each BY is equipped with an undervoltage trip coil, which de-energizes upon receipt of a trip signal from the opposite train RPS, thereby opening the BY and tripping the reactor. Unlike the RTB, the BY is not furnished with a diverse shunt trip attachment.

The RTB and BY trip logic is shown in the figure below.



Rx TRIP BREAKER A AND BYPASS BREAKER TRIP LOGICS

2.2 Current Technical Specifications Requirements

The current TS Table 3.7-1, Item 18, requires two RTBs to be operable. If one of the two RTBs becomes inoperable while the reactor is critical or at power operation, the associated TS Table 3.7-1 Action 8.A requires the unit to be in at least Hot Shutdown within 6 hours. For conditions of operation other than power operation or reactor critical, an inoperable RTB must be restored within 48 hours, or the RTBs must be opened within the next hour.

2.3 Reason for the Proposed Change

The 24-hour CT to restore an inoperable RTB will provide additional time to complete testing and maintenance while at power, thereby reducing the potential to challenge plant systems due to unnecessary transients and shutdowns associated with TS compliance. The additional 24 hours to restore an inoperable RTB also provides consistency with the established CT of 24 hours for the automatic trip logics allowed by TS Table 3.7-1, Item 19, Action 11.

2.4 Description of the Proposed Change

For conditions where the reactor is critical or at power operation, the proposed change revises Action 8.A associated with Item 18 in TS Table 3.7-1 to restore an inoperable RTB to operable status within 24 hours or be in at least Hot Shutdown within 6 hours. Additionally, a reference to WCAP-15376-P-A is added in Action 8.A.

2.5 Technical Background

The Westinghouse Owners Group (WOG – now called the Pressurized Water Reactor Owners Group (PWROG)) evaluated the change proposed by this LAR as part of an overall program addressing TS improvements for the RPS, which includes reactor trip signals and engineered safety features actuation signals.

Westinghouse topical report WCAP-15376 provides justification for increasing the CT for RTBs at Westinghouse plants. The WCAP was accepted by the NRC in a Safety Evaluation (SE) for WOG Project No. 694 on December 20, 2002. Based on NRC acceptance, as stated in TSTF-411, new test intervals and CTs were incorporated into NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." WCAP-15376 considers both a solid state protection system and a relay protection system. Surry employs a relay protection system.

As stated in WCAP-15376, "The Completion Time (CT) extensions for the reactor trip breakers will provide the utilities additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker CTs." Specifically, the WCAP provides justification for extending the CT for one inoperable RTB from 1 hour to 24 hours.

WCAP-15376 states that the RPS is not a significant contributor to plant risk from initiating events or accidents considered by PRA models for Westinghouse plants. Also, the PRA model used in the analysis for WCAP-15376 is applicable to all Westinghouse plants. However, in accordance with the SE associated with WCAP-15376, each licensee must provide a plant-specific justification in order to adopt the change. Conditions and Limitations provided in Section 5.0 of the SE require that applicability of WCAP-15376 be confirmed. Refer to Section 3.4 for responses to the SE Conditions and Limitations.

WCAP-15376 uses PRA to justify changes to the TS in accordance with RG 1.174 and RG 1.177. The risk evaluation considered the three-tiered approach as presented by the NRC in RG 1.177 for the extension to the RTB CT.

3.0 PROBABILISTIC RISK ASSESSMENT

3.1 Purpose

The purpose of this assessment is to utilize the Surry PRA to evaluate the impact on core damage frequency (CDF) and large early release frequency (LERF) for the proposed change. Using guidelines prescribed in RGs 1.174 and 1.177, this assessment evaluates the risk of changing TS Table 3.7-1, Item 18, Action 8.A to allow a single RTB to be inoperable for up to 24 hours. WCAP-15376 provides the technical justification for addition of a 24-hour CT for one inoperable RTB. The NRC issued an SE on December 20, 2002 approving WCAP-15376. TSTF-411 implements the changes proposed in WCAP-15376 to NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

3.2 Introduction

The R06d PRA model allows analysis of the conditional risk at Surry when an RTB is unavailable to perform its safety related functions utilizing a detailed probabilistic assessment of risk from internal events and internal flooding hazards at power. This risk evaluation is supplemented with qualitative insights to assess shutdown, fire, seismic, and other external risks.

RG 1.177 identified a three-tiered approach for licensees to evaluate the risk associated with the proposed TS CT change. Tier 1 is an evaluation of the impact on plant risk of the proposed TS change as expressed by the change in core damage frequency (ΔCDF), the incremental conditional core damage probability (ICCDP), the change in large early release frequency (ΔLERF), and the incremental conditional large early release probability (ICLERP). Tier 2 is an identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously or other risk-significant operational factors, such as concurrent system or equipment testing, were also involved. The objective of this part of the evaluation is to ensure that appropriate restrictions on dominant risk-significant configurations associated with the change are in place. Tier 3 is the establishment of an overall configuration risk management program (CRMP) to ensure that other potentially lower probability, but nonetheless risk-significant, configurations resulting from maintenance and other operational activities are identified and compensated for.

3.3 Analysis

Inputs

The following inputs are used for this assessment:

- Surry Average Maintenance PRA Model R06d

- Regulatory Guide 1.174
- Regulatory Guide 1.177
- CAFTA Code Suite

Risk Impact Evaluation

The NRC has identified a three-tiered approach for licensees to evaluate the risk associated with proposed TS CT changes. The following sections document the three-tiered evaluation for the addition of a 24-hour CT for one inoperable RTB

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RG 1.177 PRA Quality Evaluation

RG 1.177 contains the following discussion of PRA Technical Adequacy:

The technical adequacy of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the technical adequacy of the PRA. This applies to Tier 1 (above), and it also applies to Tier 2 and Tier 3 to the extent that a PRA model is used.

Regulatory Guide 1.200 describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for light-water reactors.

A detailed discussion and evaluation of PRA quality of the R06d model with respect to this application is provided in Attachment 4.

RG 1.177 Tier 1 Analysis

RG 1.177 contains the following discussion concerning Tier 1 Analysis:

In Tier 1, the licensee should assess the impact of the proposed TS change on CDF, ICCDP, LERF, and ICLERP. To support this assessment, two aspects need to be considered: (1) the validity of the PRA and (2) the PRA insights and findings. The licensee should demonstrate that its PRA is valid for assessing the proposed TS changes and identify the impact of the TS change on plant risk.

TS conditions addressed by CTs are entered infrequently and are temporary by their very nature. However, TS do not typically restrict the frequency of entry into conditions addressed by CTs. Therefore, the following TS acceptance guidelines specific to permanent CT changes are provided for evaluating the risk associated with the revised

CT, in addition to those acceptance guidelines given in Regulatory Guide 1.174.

The licensee has demonstrated that the TS CT change has only a small quantitative impact on plant risk. An ICCDP of less than $1.0x10^{-6}$ and an ICLERP of less than $1.0x10^{-7}$ are considered small for a single TS condition entry. (Tier 1).

RG 1.174 Acceptance Criteria are as follows:

- *When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF (Region III).*
- *When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year (Region II).*
- *Applications that result in increases to CDF above 10^{-5} per reactor year (Region I) would not normally be considered.*

Acceptance criteria for LERF are structured similarly at an order of magnitude less (1E-7, etc.).

Tier 1 Analysis Assumptions

- The PRA model R06d is valid for performing this assessment, as demonstrated in the PRA Quality Evaluation.
- The unavailability probability in the PRA model is valid based on the highest average unavailability among the four RTBs since June 2016.
- For the purposes of ΔCDF and ΔLERF calculations, this assessment assumes that overall annual RTB unavailability will increase by 100% (factor of 2) plus 24 hour unavailability as a result of the proposed change. Therefore, the total increase of failure probability is 6 times the base case. This overall increase is bounded in the analysis.
- For the purposes of ICCDP/ICLERP calculations, RTB "A" may be used as a surrogate for RTB "B" because of symmetry in the plant configuration.

Tier 1 Analysis Results

RTBs are explicitly modeled in the average maintenance model R06d. Therefore, a ΔCDF and ΔLERF for the proposed change to CTs for the RTBs may be directly calculated from the model by calculating CDF/LERF results with increased unavailability on the RTBs. Incremental core damage and large early release probabilities for a single 24-hour period with a single RTB unavailable are also calculated.

Table 1: RG 1.177 Tier 1 Analysis Results

	U1 ΔCDF	U2 ΔCDF	RG 1.177 ΔCDF Criteria	U1 ΔLERF	U2 ΔLERF	RG 1.177 ΔLERF Criteria
Reactor Trip Breaker Unavailability	6.93E-09	6.72E-09	1.00E-06	4.30E-11	4.70E-11	1.00E-07
	U1 ICCDP	**U2 ICCDP**	**RG 1.177 ICCDP Criteria**	**U1 ICLERP**	**U2 ICLERP**	**RG 1.177 ICLERP Criteria**
Single 24-hour TS Entry	1.21E-09	1.17E-09	1.00E-06	8.07E-12	8.46E-12	1.00E-07

Extensive margin exists to RG 1.177 acceptance criteria. The model does not credit any operator actions to recover failed RTBs; hence, there is no reliance on post-accident recovery of failed components affected by the CT. The dominant core damage sequences for Surry involving failure of RPS are sequences where reactor trip is required, but an ATWS occurs. Total failure to shut down the reactor occurs due to a combination of RPS failure and failure of the AMSAC System. The dominant large early release sequences for Surry involving failure of the RPS system are steam generator tube rupture (SGTR) sequences where an ATWS occurs. Total failure to shut down the reactor occurs due to a combination of RPS failure and failure of the AMSAC System.

Shutdown Risk Evaluation

The proposed change is to establish a 24-hour CT for one inoperable RTB during power operation. While the plant is shutdown, the operation of opening the RTB is no longer applicable since the RTBs must be opened to shut down the reactor. There is no shutdown risk impact on CDF or LERF associated with this CT extension. RTBs do not affect any applications on shutdown defense-in-depth or shutdown key safety functions.

Internal Fire Hazard Evaluation

There is currently no Fire PRA model for Surry. The Surry Individual Plant Examination of Non-seismic External Events and Fires (IPEEE) identified four areas as significant contributors to the fire CDF: the Cable Vault and Tunnel (CVT), the Emergency Switchgear Room (ESGR), the Main Control Room (MCR), and the Normal Switchgear Room (NSGR). Since fires in other areas, such as the cable spreading room where the reactor trip breakers are located, are not significant contributors to fire risk as characterized by the IPEEE, they are screened from further consideration.

Due to the fail-safe design and reliability of the normal reactor trip function, the availability of the AMSAC System, which can shutdown the reactor if an RTB failed to open, and the reliability of operator action to trip the reactor, an increase in risk from fires associated with this proposed CT extension is assessed to be negligible.

Seismic Hazard Evaluation

There is currently no PRA Standard Compliant Seismic PRA model for Surry. The Surry Seismic Probability Risk Assessment (SPRA) Pilot Plant Report was reviewed, and this proposed change does not impact the conclusions drawn in that report.

Other significant sequences from the Surry SPRA Pilot Plant Report involving seismic induced transients and loss of offsite power events were reviewed, and it was determined that the RTB out-of-service-time does not play a significant role in these sequences. Therefore, it is concluded that the proposed change has a negligible impact on seismic risk and is screened from further evaluation. Discussion to support this evaluation is provided in the following paragraphs.

Generic Letter (GL) 88-20, Supplement 4, was issued by the NRC in June 1991. This letter and NUREG-1407 requested each nuclear plant licensee to perform an IPEEE. In a December 1991 letter to the NRC, Surry identified the planned approach to address the IPEEE. For non-seismic external events and fires, the IPEEE effort was completed and a report was submitted to the NRC in December 1997.

Surry was categorized in NUREG-1407 as a focused scope plant. As identified in Surry's December 1991 letter, the Seismic Margins Method (SMM) developed by Electric Power Research Institute (EPRI) with enhancements was selected for Surry. A completion schedule for IPEEE - Seismic was initially provided by Surry in its September 1992 letter to the NRC which also noted that elements of the effort to resolve IPEEE - Seismic, notably plant walkdowns, would be integrated with the resolution of Unresolved Safety Issue (USI) A-46 identified in NRC GL 87-02, Supplement 1, issued in May 1992.

In September 1995, the NRC issued Supplement 5 to GL 88-20. This letter gave further guidance on the basis for selection of components that needed capacity evaluation. Based on GL 88-20, Supplement 5, Surry submitted a revised approach to the NRC in November 1995. This approach, while still retaining the SPRA methodology and treating Surry as a focused scope plant, identified areas where screening and judgment by experienced and trained engineers would eliminate the need for performing capacity calculations for rugged components, structures, and systems, and required such evaluations only for weaker and critical components. The IPEEE - Seismic program at Surry was performed in accordance with the SPRA methodology for a focused plant and Surry's stated commitments.

In 2010, EPRI published TR-1020756 which documents a pilot study of seismic risk for Surry. The pilot study used the Surry SPRA developed for IPEEE as a starting point and updated the elements of the SPRA in conformance with the project team's interpretation of the Capability Category II requirements in the new ASME/ANS PRA Standard. This resulted in the development of a new seismic hazard for the Surry site, calculation of the site response to the new hazard, calculation of a new structural response, development of new seismic fragilities, and quantification of the seismic risk based on updated seismic systems logic models. This SPRA model has not been peer reviewed against the PRA Standard, but still represents the most accurate understanding of seismic risk for Surry.

The SPRA Pilot Plant Report identified ATWS as a very small contributor to seismic risk for Surry. Due to the fail-safe design and reliability of the RPS, along with operator actions to shut down the reactor following a seismic event, the seismic risk associated with this proposed change is assessed to be negligible.

Other External Hazards Evaluation

Other external events have been screened from further evaluation for the proposed change based on the following discussion.

Other external hazards, as identified by NUREG/CR-2300, "A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," and NUREG/CR-4839, "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," have been taken into consideration. These hazards were evaluated in response to GL 88-20, Supplement 4. The analysis utilized the method and results obtained by NUREG/CR-4550, "Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines, Vol 1," supplemented with information from the Surry UFSAR. All other external events, except for external flooding, aircraft impacts, and pipeline accidents, were screened out based on the UFSAR and NUREG/CR-4550 information. A bounding analysis, based on the methods used by NUREG/CR-4550, was performed for the effects of aircraft impacts and pipeline accidents, and these initiators were shown to have a small frequency of occurring and are, therefore, screened from further consideration. External flooding was evaluated separately and also shown to be a non-significant contributor to overall core damage risk at Surry. Therefore, it can be concluded that non-seismic and non-fire external events are not significant risk contributors to this evaluation.

RG 1.177 Tier 2: Avoidance of Risk Significant Plant Configurations

RG 1.177 contains the following discussion concerning Tier 2 analysis:

The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change. An effective way to perform such an assessment is to evaluate equipment according to its contribution to plant risk (or safety) while the equipment covered by the proposed CT change is out of service. Evaluation of such combinations of equipment out of service against the Tier 1 ICCDP and ICLERP acceptance guidelines could be one appropriate method of identifying risk-significant configurations. Once plant equipment is so evaluated, an assessment can be made as whether certain enhancements to the TS or procedures are needed to avoid risk significant plant configurations. In addition, compensatory actions that can mitigate any corresponding increase in risk (e.g., backup equipment, increased surveillance frequency, or upgraded procedures and training) should be identified and evaluated. Any changes made to the plant design or operating procedures as a result of such a risk evaluation (e.g., required backup equipment, increased surveillance frequency, or upgraded procedures and training required before certain plant system configurations can be entered) should be incorporated into the analyses utilized for TS changes as described under Tier 1 above.

A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 PRA model did not reveal any risk significant maintenance configurations when one RTB is unavailable. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable. [Regulatory Commitment] No additional enhancements, procedure revisions or compensatory actions are recommended from the Tier 2 evaluation.

RG 1.177 Tier 3: Risk-Informed Plant Configuration Control and Management

Dominion Energy Virginia's 10 CFR 50.65(a)(4) program fully satisfies the recommendations of RG 1.177, Tier 3. RG 1.177, Section 2.3 states that:

The licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. A viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation.

Dominion Energy Virginia's 10 CFR 50.65(a)(4) program performs PRA analyses of planned maintenance configurations in advance. The RTBs are included in the 10 CFR 50.65(a)(4) scope and their removal from service is monitored, analyzed, and managed. Configurations that approach or exceed the NUMARC 93-01 risk limits are identified and either avoided or addressed by risk management actions. Emergent configurations

are identified and analyzed by the on-shift staff for prompt determination of whether risk management actions are needed. The configuration analysis and risk management processes are fully proceduralized in compliance with the requirements of 10 CFR 50.65(a)(4). Dominion Energy Virginia's (a)(4) program is implemented by station procedures.

To support Dominion Energy Virginia's 10 CFR 50.65(a)(4) program, a dedicated PRA model is used to perform configuration risk analysis. The model uses the SPS-R06b model as a framework with some adjustments to optimize the model for configuration risk calculations. The model allows for quantitative Level 1 and Level 2 (LERF) assessments of internal events and internal floods hazards for at-power configurations. Risk during shutdown configurations and risks due to other hazards are assessed qualitatively. Changes in plant configuration or PRA model features are dispositioned and managed by Dominion Energy Virginia's PRA configuration control process. Procedures are in place to ensure that actions are taken as necessary to qualitatively assess configurations outside the scope of the PRA model.

3.4 Conditions and Limitations from NRC Safety Evaluation for WCAP-15376

Westinghouse topical report WCAP-15376 provides justification to increase the CT for RTBs at Westinghouse plants. Section 5.0 of the NRC SE that approved WCAP-15376 stated that the applicability of WCAP-15376 on a plant-specific basis needs to be confirmed by providing the following information:

1. *A licensee is expected to confirm the applicability of the topical report to their plant, and to perform a plant-specific assessment of containment failures and address any design or performance differences that may affect the proposed changes.*

   Surry Response: The applicability of WCAP-15376 to Surry has been assessed and confirmed. The RPS/ESFAS functions at Surry are similar in design and response to the topical report reference plant. Plant-specific evaluations of current and future unavailability were taken into account in the Tier 1 analysis.

   A plant specific assessment of containment failures and other failure modes which could result in a release was conducted to support the development the Surry LERF model. This plant-specific model addressed the specific design and performance of Surry containment. The LERF model was used to evaluate the proposed change, and the resulting estimated increase in LERF was very small.

2. *Address the Tier 2 and Tier 3 analyses including risk significant configuration insights and confirm that these insights are incorporated into the plant-specific configuration risk management program.*

Surry Response: Tier 2 analysis was performed for Surry. A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 results was performed, and no risk significant maintenance configurations were identified that must be avoided when one RTB is out of service. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable.

Dominion Energy Virginia's 10 CFR 50.65(a)(4) program fully satisfies the recommendations of RG 1.177, Tier 3. As stated in the Reg Guide 1.177 Tier 1 Discussion in section 3.3, the 10 CFR 50.65(a)(4) program performs PRA analyses of planned maintenance configurations in advance. The RTBs are included in the 10 CFR 50.65(a)(4) scope and their removal from service is monitored, analyzed, and managed. To support the 10 CFR 50.65(a)(4) program, a dedicated PRA model is used to perform configuration risk analysis. The model uses the SPS-R06b model as a framework with some adjustments to optimize the model for configuration risk calculations. The model allows for quantitative Level 1 and Level 2 (LERF) assessments of internal events and internal floods hazards for at-power configurations. Risk during shutdown configurations and risks due to other hazards are assessed qualitatively. Changes in plant configuration or PRA model features are dispositioned and managed by Dominion Energy Virginia's PRA configuration control process. Procedures are in place to ensure that actions are taken, as necessary, to qualitatively assess configurations outside the scope of the PRA model. The 10 CFR 50.65(a)(4) program will be used to ensure high risk configurations are not entered when one RTB is out of service and risk management actions are applied, as needed.

3. *The risk impact of concurrent testing of one logic cabinet and associated reactor trip breaker needs to be evaluated on a plant-specfic basis to ensure conformance with the WCAP-15376 evaluation, and RGs 1.174 and 1.177 guidance.*

   Surry Response: The plant-specific evaluation performed for Surry assessed the risk with one RTB unavailable concurrent with one train of RPS unavailable. The results conform with the RG 1.174 and RG 1.177 guidance.

4. *To ensure consistency with the reference plant, the model assumptions for human reliability in WCAP-15376 should be confirmed to be applicable to the plant-specific configuration.*

   Surry Response: The assumptions for the following operator actions were reviewed:

Human Error Probability (HEP) Reliability in WCAP-15376	Equivalent HEP Description in Surry R06d Model	HEP Term in Surry R06d Model
Reactor trip from the main control board trip switches	1/2-E-0 - OPERATOR FAILS TO MANUALLY TRIP REACTOR	HEP-C-TRIPRX
Reactor trip by interrupting power from the motor-generator sets given that the operator failed to trip by the control board switches	1/2-E-0 - OPERATOR FAILS TO MANUALLY TRIP MG SETS	HEP-C-TRIPMG
Manually insert the control rods into the core given the previous operator actions to trip have failed	1/2-FR-S.1 - OPERATOR FAILS TO INITIATE ROD INSERTION	HEP-C-INSERT
Safety injection from the main control board switches	1-E-0 - OPER FAILS TO MANUALLY INITIATE U1 SI	HEP-C-1MANSI
Safety injection by manual actuations of individual components		
Auxiliary feedwater pump start	No credit taken for manual actuation if Aux Feedwater pump fails to start	N/A

The assumptions used in the Surry plant-specific analysis were confirmed to be either consistent with or more conservative than the HEP assumptions of WCAP-15376.

5. *For future digital upgrades with increased scope, integration and architectural differences beyond that of Eagle 21, the staff finds the generic applicability of WCAP-15376 to future digital systems not clear and should be considered on a plant-specific basis.*

   Surry Response: There are no digital components within the RTBs or the associated logic cabinets. Therefore, this item is not applicable for Surry.

3.5 Results and Conclusions

The increase in risk associated with the proposed TS change is consistent with the RG 1.174 and RG 1.177 acceptance guidelines for a permanent TS CT change. This evaluation demonstrates that nuclear defense-in-depth will not be significantly impacted

by allowing a single RTB to be unavailable for up to 24 hours. A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 PRA model did not reveal any risk significant maintenance configurations when one RTB is unavailable. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable. No additional enhancements, procedure revisions or compensatory actions are recommended from the Tier 2 evaluation.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

The regulations in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establish minimum principal design criteria for water-cooled nuclear power plants, while 10 CFR 50 Appendix B and the licensee quality assurance programs establish quality assurance requirements for the design, manufacture, construction, and operation of structures, systems, and components. The current regulatory requirements of 10 CFR 50 Appendix A that are applicable to the Reactor Protection System (RPS) include: General Design Criteria (GDC) 1, 20, 21, 22, and 23.

During the initial plant licensing of Surry Units 1 and 2, it was demonstrated that the design of the RPS met the regulatory requirements in place at that time. The GDC included in Appendix A to 10 CFR 50 did not become effective until May 21, 1971. The Construction Permits for Surry Units 1 and 2 were issued prior to May 21, 1971; consequently, Surry Units 1 and 2 were not subject to current GDC requirements (SECY-92-223, dated September 18, 1992). The following information demonstrates Surry Units 1 and 2 meet the intent of the GDC published in 1967 (Draft GDC). Specifically, Section 1.4 of the Surry Updated Final Safety Analysis Report (UFSAR) discusses Surry compliance with these criteria. The draft GDC associated with the RPS are addressed below.

- Quality Standards (Criterion 1 - draft)

  Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected in accordance with quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. A showing of sufficiency and applicability of codes, standards, quality

assurance programs, test procedures, and inspection acceptance levels used is required.

Design Conformance

Structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The Quality Assurance Program was established to provide assurance that safety-related structures, systems, and components satisfactorily perform their intended safety functions.

- Core Protection Systems (Criterion 14 - draft)

  Core protection systems, together with associated equipment, are designed to prevent or to suppress conditions that can result in exceeding acceptable fuel damage limits.

  Design Conformance

  The reactor protection system receives, from unit instrumentation, signals that are indicative of an approach to an unsafe operating condition. This system then actuates alarms, prevents control rod assembly motion, initiates load runback, and/or opens the trip breakers causing the insertion of the control rod assemblies, depending on the severity of the condition. The allowable operating range within reactor trip settings includes combinations of power, temperature, and pressure that do not result in the occurrence of a departure from nucleate boiling with all reactor coolant pumps in operation.

- Protection Systems Reliability (Criterion 19 - draft)

  Protection systems are designed for high functional reliability and inservice testability necessary to avoid undue risk to the health and safety of the public.

  Design Conformance

  The reactors use the Westinghouse magnetic-type control rod drive mechanisms that are similar to those used in the San Onofre, Indian Point, and Connecticut Yankee power stations. Upon a loss of power to the coils, the control rod assembly is released and falls by gravity into the core. The reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and testing at power. The bypass removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; that is, a two-out-of-three circuit becomes

a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel. Reliability and independence are obtained by redundancy within each tripping function.

- Protection Systems Redundancy and Independence (Criterion 20 - draft)

  Redundancy and independence designed into the protection systems are sufficient to ensure that no single failure or removal from service of any component or channel of such a system results in a loss of the protection function. The redundancy provided includes, as a minimum, two channels of protection for each protection function to be served.

  Design Conformance

  The reactor protection system is designed in accordance with the IEEE *Standards for Nuclear Power Plant Protection Systems*. Two reactor trip breakers are provided to interrupt power to the control rod drive mechanisms. The main breaker contacts are connected in series with the mechanism coils. Opening either breaker interrupts power to all mechanisms, causing them to release all control rod assemblies to fall by gravity into the core. Each breaker is opened through an undervoltage trip coil. A shunt trip relay is installed in parallel with the undervoltage attachment. Upon de-energization, contacts from the relay energize the reactor trip breaker shunt trip attachment and trips open the breaker. This provides a redundant/backup means to automatically trip the breakers upon the receipt of a trip signal from the reactor trip system. Each protection channel permits the actuation of one reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of power to the control rod drive mechanisms.

- Demonstration of Functional Operability of Protection Systems (Criterion 25 – draft)

  Means shall be included for the suitable testing of the active components of protection systems while the reactor is in operation to determine if a failure or loss of redundancy has occurred.

  Design Conformance

  Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation. This includes a check through to the trip breakers that includes the trip logic. Thus, the operability of each trip channel is determined conveniently and without ambiguity.

- Protection Systems Fail-Safe Design (Criterion 26 - draft)

  The protection systems are designed to fail into the safe state or into a state established as tolerable on a defined basis if conditions such as a disconnection of the system, a loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

  Design Conformance

  Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized. An open circuit or loss of channel power therefore causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from two independent electrical buses. Failure to de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

  Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on each drive, allowing the control rod assemblies to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

Quality Assurance

Quality assurance criteria provided in 10 CFR Part 50, Appendix B, that apply to the RTS include: Criteria III, V, XI, XVI, and XVII. Criteria III and V require measures to ensure that applicable regulatory requirements and the design basis, as defined in 10CFR50.2, "Definitions," and as specified in the license application, are correctly translated into controlled specifications, drawings, procedures, and instructions. Criterion XI requires a test program to ensure that the subject systems will perform satisfactorily in service and requires that test results shall be documented and evaluated to ensure that test requirements have been satisfied. Criterion XVI requires measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected, and that significant conditions adverse to quality are documented and reported to management. Criterion XVII requires maintenance of records of activities affecting quality.

4.2 No Significant Hazards Consideration

The proposed change revises Action 8.A associated with Item 18 in Surry Power Station Technical Specifications (TS) Table 3.7-1, "Instrumentation Systems," for one inoperable Reactor Trip Breaker (RTB). The current Action requires the plant to be in at least Hot Shutdown within 6 hours if one RTB is inoperable. The revised Action provides a CT of 24 hours to restore an RTB to operable status. The 24-hour CT

provides additional time to perform maintenance activities at power while minimizing the risk associated with the loss of the component function. The proposed change reduces the potential for unnecessary shutdowns associated with TS compliance.

The proposed 24-hour CT to restore an inoperable RTB to operable status was approved by the NRC for Westinghouse plants in Westinghouse topical report, WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," Revision 1 (hereafter referred to as WCAP-15376). Based on NRC acceptance as stated in TSTF-411, "Surveillance Test Interval Extensions for Components of the Reactor Protection System," Revision 1 (hereafter referred to as TSTF-411), new test intervals and CTs were incorporated into NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4. This license amendment request (LAR) uses WCAP-15376 and the Surry-specific probabilistic risk assessment (PRA) as justification for the proposed CT extension.

The supporting PRA performed for the CT extension concluded that the increase in risk associated with the proposed change is consistent with acceptance guidelines for a permanent CT change contained in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Bases," dated July 1998, and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," dated August 1998. The supporting PRA demonstrates that defense-in-depth will not be significantly impacted by allowing a single RTB to be inoperable for up to 24 hours.

Dominion Energy Virginia has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change provides a 24-hour CT for restoration of an inoperable RTB. RPS performance will remain within the bounds of the previously performed accident analyses since no change to reactor trip instrumentation or plant hardware is being made. The RPS will continue to function in a manner consistent with the plant design basis.

   The proposed change does not modify any system interfaces and does not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment

assumed to function during an accident situation. There is no change to normal plant operating parameters or accident mitigation performance.

The determination that the results of the proposed change are acceptable was established in the NRC Safety Evaluation (SE) prepared for WCAP-15376-P-A. Implementation of the proposed change will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the NRC SE conditions.

The proposed change to add the CT reduces the potential for unnecessary entries into TS action statements and resultant plant transients and, therefore, does not increase the probability of any accident previously evaluated. The proposed change does not alter the response of the plant to any accidents. The RPS instrumentation and RTBs will remain highly reliable, and the proposed changes will not result in a significant increase to the risk of plant operation. The PRA performed for the proposed CT change is based on justification presented in NRC approved WCAP-15376. The PRA concluded that the increase in risk associated with the proposed change is consistent with the RG 1.174 and RG 1.177 acceptance guidelines for a permanent TS CT change. The PRA demonstrates that defense-in-depth will not be significantly impacted by allowing a single RTB to be inoperable for up to 24 hours.

A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 PRA model did not reveal any risk significant maintenance configurations when one RTB is unavailable. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable. No additional enhancements, procedure revisions or compensatory actions are recommended from the Tier 2 evaluation.

Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change provides a 24-hour CT for restoration of an inoperable RTB. There are no hardware changes, nor are there any changes in the method by which any safety related plant system performs its safety function. The proposed change does not affect the normal method of plant operation and does not result in physical alteration to any plant system. The proposed change does not include any changes to instrumentation setpoints or changes to accident analysis assumptions. No new accident scenarios, transient precursors, failure mechanisms, or limiting single

failures are introduced as a result of this change. There will be no adverse effects or challenges imposed on any safety-related system as a result of the proposed change.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides a 24-hour CT for restoration of an inoperable RTB. The proposed change does not adversely affect any current plant safety margins or the reliability of equipment assumed in the safety analysis. There are no changes being made to any safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety. Furthermore, as noted above, a supporting PRA was performed for the proposed CT. The PRA concluded that the increase in risk associated with the proposed change is consistent with the RG 1.174 and RG 1.177 acceptance guidelines for a permanent TS CT change. This PRA demonstrates that defense-in-depth will not be significantly impacted by allowing a single RTB to be inoperable for up to 24 hours.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Dominion Energy Virginia concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Precedents

The following plants have made submittals proposing changes similar to those being proposed in this LAR. The submittals below included a 24-hour RTB completion time and were approved by the NRC.

- Donald C. Cook - submitted on August 30, 2002 and approved on May 23, 2003 by Amendments 277 (Unit 1) and 260 (Unit 2). [ML031320614]
- Callaway - submitted on December 17, 2003 and approved on January 31, 2005 by Amendment 165. [ML050320484]
- Vogtle - submitted on January 27, 2005 and approved on September 1, 2006 by Amendments 145 (Unit 1) and 125 (Unit 2). [ML062360587]
- Beaver Valley - submitted on December 21, 2007 and approved on December 29, 2008 by Amendments 282 (Unit 1) and 166 (Unit 2). [ML083380061]

- Millstone Unit 3 - submitted May 8, 2014 and approved November 30, 2015 by Amendment 266. [ML15288A004]
- Virgil C. Summer - submitted December 16, 2015 and approved October 4, 2017 by Amendment 209. [ML17206A412]

5.0 ENVIRONMENTAL ASSESSMENT

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REGULATORY COMMITMENT

A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 PRA model did not reveal any risk significant maintenance configurations when one RTB is unavailable. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable.

7.0 CONCLUSION

The proposed change revises Action 8.A associated with Item 18 in Surry Power Station TS Table 3.7-1, titled "Reactor Trip Instrument Operating Conditions." The proposed change provides a CT to restore one of the two RTBs to operable status within 24 hours or be in at least Hot Shutdown within 6 hours. The CT provides additional time to perform maintenance activities at power and enhances safety by reducing the potential for plant shutdowns. The proposed CT does not impact the design function of the RTBs. Additionally, the proposed change does not physically alter plant equipment, does not impact plant operation, and does not affect the safety analyses.

The basis for the proposed change is NRC-approved Westinghouse Topical Report WCAP-15376. In addition, a supporting PRA was performed for the proposed CT extension. The PRA concluded that the increase in risk associated with the proposed

CT extension is consistent with the RG 1.174 and RG 1.177 acceptance guidelines for a permanent TS completion time change. The PRA demonstrates that defense-in-depth will not be significantly impacted by allowing a single RTB to be unavailable for up to 24 hours. A detailed review of PRA importance metrics (Risk Achievement Worth, Fussell-Vesely) from the Tier 1 PRA model did not reveal any risk significant maintenance configurations when one RTB is unavailable. To maintain appropriate measures of defense in depth, no maintenance will be planned on the AMSAC system while one RTB is inoperable. No additional enhancements, procedure revisions or compensatory actions are recommended from the Tier 2 evaluation.

Therefore, Dominion Energy Virginia concludes, based on consideration discussed herein, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change, (2) such activities will be conducted in accordance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment 2

MARKED-UP TECHNICAL SPECIFICATIONS PAGE

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

restore the inoperable channel to OPERABLE status within 24 hours or

If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours.

ACTION 8.A. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE, or one reactor trip breaker may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor trip breaker and automatic trip logic provided the other train is OPERABLE.

(Reference: WCAP-15376-P-A).

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

Attachment 3

PROPOSED TECHNICAL SPECIFICATIONS PAGE

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.
2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours.

ACTION 8.A. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within 6 hours (Reference: WCAP-15376-P-A). In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE, or one reactor trip breaker may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor trip breaker and automatic trip logic provided the other train is OPERABLE.

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required

Attachment 4

PROBABILISTIC RISK ASSESSMENT QUALITY EVALUATION

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Station Units 1 and 2

PROBABILISTIC RISK ASSESSMENT (PRA) QUALITY EVALUATION

The Surry internal events PRA model was originally developed in 1991 for the Individual Plant Examination (IPE) submittal. Since then, the Surry PRA model has been maintained and upgraded to ensure the model reflects the as-built, as operated plant and to improve the accuracy of risk assessment. The current Surry PRA model which was used to support this analysis is highly detailed and has broadened from the original scope to accurately assess plant Core Damage Frequency (CDF) and Large, Early Release Frequency (LERF) from internal events and internal floods. Since the IPE, the Surry PRA model has benefited from the following technical peer reviews:

History of Surry PRA Model Technical Peer Reviews

Technical Review	Review Year	PRA Standard	RG 1.200 Revision
WOG Peer Review	1998	Pre-NEI-00-02 pilot using BWROG-97026	N/A
Scientech HRA Peer review	2004	RA-Sa-2003	Trial Use
Self-Assessment	2007	RA-Sb-2005	1
Westinghouse Peer Focused Scope Peer Review (FSPR)	2010	RA-Sa-2009	2
Westinghouse Peer FSPR	2012	RA-Sa-2009	2
Westinghouse Peer FSPR	2016	RA-Sa-2009	2
EPM FSPR (March)	2018	RA-Sa-2009	2
EPM F&O Closure Assessment (March)	2018	RA-Sa-2009	2
EPM FSPR (October)	2018	RA-Sa-2009	2

Since 2010, industry peer reviews have evaluated the Surry model against the Supporting Requirements (SRs) in PRA Standard RA-Sa-2009 and RG 1.200. The current Surry PRA internal events model has been determined to meet or exceed Capability Category II requirements for the SRs in RA-Sa-2009 and Regulatory Guide (RG) 1.200.

In addition to the peer reviews described in the table above, a review was performed in March 2018 to close out existing facts and observations (F&Os) by Independent Assessment using the process described in NEI 05-04, Appendix X. The requirements described in Section X.1.3 associated with selection of the Independent Assessment team members, preparation and documentation by the host utility, conduct of the review, and documentation in a final report were met. Sixty-one F&Os were submitted to the Independent Assessment team for closure, and the 61 F&Os were deemed closed.

There are six F&Os on the Surry internal events PRA model that are currently open. One was carried over from the February 2017 Peer Review, one was carried over from the May 2018 Peer Review, and four are from the October 2018 Peer Review. These F&Os and their impact on this application are described in the table below.

Open F&Os for Surry Internal Events PRA Model SPS-R06d

Peer Review	F&O #	Summary of Finding	Impact of F&O on Proposed Change
Feb 2017 Peer Review	QU-F2-01	Dominion Energy's PRA update process periodically creates a new "model of record" that addresses the requirements of QU-F2 & QU-F3. However interim updates are performed to maintain the PRA consistent with the as-built/as-operated plant that does not address all of requirements of QU-F2 & QU-F3. This focused scope peer review (FSPR) reviewed interim model MC.1, which did not include the comprehensive results analysis such as, but not limited to, the truncation level sensitivity study required to meet the standard.	The full set of sensitivity studies performed to meet all QU requirements is not repeated for every Interim PRA model change; however, the results of every interim model change are reviewed in detail to ensure the new results are appropriate and reasonable given the changes applied. Repeating all sensitivity studies related to QU requirements is not expected to impact any of the risk insights generated in this analysis, so this finding has no impact on the acceptability of this application.
May 2018 Peer Review	19-7	Success criteria for MSLB include the impact of failure to isolate AFW to the faulted SG with a small probability of vessel failure. The more likely impact of failure to isolate AFW is more rapid depletion of CST inventory. This should be included in the MSLB sequences.	This F&O was resolved in the model used to perform this analysis. The impact of main steam isolation failures on AFW was explicitly modeled for MSLB sequences.

Peer Review	F&O #	Summary of Finding	Impact of F&O on Proposed Change
October 2018 Peer Review	HR-A1-01	The SR states "for equipment modeled in the PRA, IDENTIFY activities ...". The system notebooks contain: (1) a list of test and maintenance procedures that are "modeled as potential pre-initiator human error events", and (2) a list of other test and maintenance procedures that "do not involve pre-initiator human error events". It is not clear what the bases are for selection of the procedures to be modeled as pre-initiators. Attachment 8 to the system notebooks includes a tab "T&M Unavailability BEs" that lists test and maintenance unavailability events, and a tab "Human Actions" listing the pre-initiators defined for the system, but there is no documented link between procedure, T&M event, and pre-initiator HFE.	A detailed review of test and maintenance procedures was performed to determine which pre-initiator human error events should be included in the model. Improving the documentation to clarify the link between procedures, T&M events, and pre-initiator HFEs will not impact the model quantification, so this finding has no impact on the acceptability of this analysis.
October 2018 Peer Review	HR-B1-01	Screening criteria that can be used to screen components/failure modes from further consideration for preinitiators are provided in Attachment 2 of NF-AA-PRA 101-2051, Revision 4. However, the screening criteria applied to the screening of procedures and activities are not documented. In Attachment 8 of the system notebooks, maintenance unavailability events are listed in the "T&M Unavailability Bes". Each such activity should have a corresponding preinitiator HFE, unless it can be screened out. When comparing the list of T&M events to the list in the "Human Actions" tab, it is not clear on what bases T&M activities were screened out, and it is not clear which HFEs relate to which activities.	Additional documentation clarifying the bases on how T&M events were screened out from inclusion as pre-initiator human error events will not impact model quantification, so this finding has no impact on the acceptability of this analysis.

Peer Review	F&O #	Summary of Finding	Impact of F&O on Proposed Change
October 2018 Peer Review	IE-A6-01	Common cause failures of CW and SW components are not included in the initiating event fault trees. Evaluate the inclusion of initiating event common causes, such as pump failures and traveling screen plugging, in the LOCW event tree. Include the events or document the basis for exclusion.	Common cause failures of CW and SW components are included in the Surry PRA model. For a cooling water support system initiating event (SSIE) to take place, a single failure is evaluated over a mission time of one year and combined with failure modes of the parallel components (including common cause) over a mission time of 24 hours. This is an industry standard method of modeling SSIEs as described in EPRI TR-1016741. Modeling a common cause failure of multiple trains of cooling water components over a mission time of one year would be excessively conservative as it would assume the first failure would be unable to be repaired during the mission time in the extensive period before the subsequent failures took place. Additional documentation justifying the current modeling of cooling water systems will not impact model quantification, so this finding has no impact on the acceptability of this analysis.
October 2018 Peer Review	SY-A11-01	SY-A11 directs the analyst to exclude components only if the quantification criteria presented in SY-A15 are met. Although some components were excluded based on the criteria in SY-A15, some components were excluded based on qualitative arguments such as "failure frequency was negligible".	Components were excluded based on qualitative or quantitative justification for not being significant contributors to system or functional reliability. If it was determined that some components excluded based on qualitative justification did not meet the quantitative criteria for exclusion, it is still expected that these components would not be significant contributors based on the qualitative bases that are documented. Inclusion of additional non-significant basic events will not have a significant impact on the model quantification, so this finding does not impact the acceptability of this analysis.

The upgrades applied to the model used for this analysis have been peer reviewed. A review of model assumptions and uncertainties did not identify any impact to this application or require additional evaluation.

Probabilistic Risk Analysis Configuration Control Review

Dominion Energy's PRA utilizes a Probabilistic Risk Analysis Configuration Control (PRACC) program that is described in procedures applicable to the Dominion Energy fleet. The PRACC program includes the following attributes:

- A process for monitoring PRA inputs and collecting new information

- A process that maintains and upgrades the PRA to be consistent with the as-built, as operated plant
- A process that ensures that the cumulative impact of pending changes is considered when applying the PRA
- Guidance for documentation of the PRA Maintenance and Upgrade process

In addition, the computer codes used to support the PRACC program and perform PRA model quantification are controlled. This PRACC process has been peer reviewed and satisfies configuration control requirements specified in Section 1-5 of the ASME/ANS PRA standard. In accordance with this process, open items in Dominion Energy's PRACC database were reviewed to ensure the cumulative impact of pending changes to the PRA is considered in this evaluation.

A review of the PRACC database shows there are no pending changes that would have a significant impact on this application or require additional evaluation.