VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	19-427
SPS-LIC/CGL	R0
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
ANNUAL SUBMITTAL OF TECHNICAL SPECIFICATIONS BASES CHANGES
PURSUANT TO TECHNICAL SPECIFICATION 6.4.J

Pursuant to Technical Specification 6.4.J, "Technical Specifications (TS) Bases Control Program," Dominion Energy Virginia hereby submits changes to the Bases of the Surry TS implemented between October 1, 2018 and September 30, 2019.

Bases changes to the TS that were not previously submitted to the NRC as part of a License Amendment Request were reviewed and approved by the Facility Safety Review Committee. It was determined that the changes did not require a revision to the TS or operating licenses, nor did the changes involve a revision to the UFSAR or Bases that required NRC prior approval pursuant to 10CFR50.59. These changes have been incorporated into the TS Bases. A summary of these changes is provided in Attachment 1.

TS Bases changes that were submitted to the NRC for information along with associated License Amendment Request transmittals, submitted pursuant to 10CFR50.90, were also reviewed and approved by the Facility Safety Review Committee. These changes have been implemented with the respective License Amendments. A summary of these changes is provided in Attachment 2.

Current TS Bases pages reflecting the changes discussed in Attachments 1 and 2 are provided in Attachment 3.

If you have any questions regarding this transmittal, please contact Mrs. Candee G. Lovett at (757) 365-2178.

Very truly yours,

Rob M. Garver II
Director Station Safety and Licensing
Surry Power Station

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NRR

Attachments:
1. Summary of TS Bases Changes Not Previously Submitted to the NRC
2. Summary of TS Bases Changes Associated with License Amendments
3. Current TS Bases Pages

Commitments made in this letter: None.

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Attachment 1

Serial No. 19-427

Summary of TS Bases Changes Not Previously Submitted to the NRC

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

SUMMARY OF TS BASES CHANGES NOT PREVIOUSLY SUBMITTED TO THE NRC

TS 3.7 Basis Revision (TS Basis Pages TS 3.7-6 and TS 3.7-6a)

A revision was made in the TS 3.7 Basis. The TS 3.7 Basis revision defines the operability requirements for the Open Phase Isolation System. This TS 3.7 Basis revision is related to the TS 3.7 and TS 4.1 revisions for Open Phase Protection approved by TS Amendments 292/292. TS Amendment --/292 was implemented on Unit 2 on November 19, 2018.

The TS 3.7 Basis change was approved on October 17, 2017 and implemented on Unit 2 on November 19, 2018.

Attachment 2

Serial No. 19-427

Summary of TS Bases Changes Associated with License Amendments

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

SUMMARY OF TS BASES CHANGES ASSOCIATED WITH LICENSE AMENDMENTS

Temporary, One-time 21-day AOT for Reserve Station Service Transformer (RSST) C Replacement (TS Bases Pages TS 3.16-7, TS 3.16-7a, and TS 3.16-7b)

These amendments revised Technical Specification 3.16, "Emergency Power System," to provide a temporary, one-time 21-day Allowed Outage Time for replacement of RSST C.

The associated Bases changes were included for information in a November 7, 2017 letter (Serial No. 17-435). These changes were incorporated into the Bases as part of the October 24, 2018 implementation of License Amendments 293/293 issued on October 5, 2018.

Revised Reactivity Limits / Adoption of TSTF-490 and Updated Alternative Source Term (AST) Analyses (TS Basis Pages TS 3.1-15b, TS 3.1-16, TS 3.1-17, TS 3.1-17a, TS 4.1-5a, and TS 4.1-5b)

These amendments included the following two items: 1) adoption of TSTF-490 to replace the primary coolant gross specific activity with limits on primary coolant noble gas activity and 2) updating the AST analyses bases for new codes, revised atmospheric dispersion factors, new fuel handling accident fuel rod gap fractions and control room isolation operator action time, and elimination of the locked rotor accident dose consequences.

The associated Basis change was included for information in a March 2, 2018 letter (Serial No. 18-069). This change was incorporated into the Basis as part of the June 26, 2019 implementation of License Amendments 295/295 issued on June 12, 2019.

Attachment 3

Serial No. 19-427

Current TS Bases Pages

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

BASES

BACKGROUND - The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The Reactor Coolant System (RCS) specific activity Limiting Condition for Operation (LCO) limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES - The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate Standard Review Plan acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.6.H, Secondary Specific Activity.

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity is at 10.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 234 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a coincident loss of offsite power. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the Turbine Building to maximize control room dose. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO - The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 234 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY - In REACTOR OPERATION conditions where Tavg exceeds 200°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In COLD SHUTDOWN and REFUELING SHUTDOWN the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

3.1.D.1.a

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is $\leq$ 10.0 μCi/gm. The completion time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

3.1.D.1.b

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required actions 3.1.D.1.a and b while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.1.c

If the required action of Condition 3.1.D.1.a or 3.1.D.1.b is not met, or if the DOSE EQUIVALENT I-131 is > 10.0 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3.1.D.2.a

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within the limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required action 3.1.D.2.a while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.2.b

If the required action or associated Allowed Outage Time of Condition 3.1.D.2.a is not met, or if the DOSE EQUIVALENT XE-133 is > 234 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required action and completion time are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. 10 CFR 50.67
2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms"
3. UFSAR, Section 14.3.1 Steam Generator Tube Rupture
4. UFSAR, Section 14.3.2 Steam Line Break

RCS Flow

This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of ≥ 90% of RATED POWER (i.e., shall be performed within 7 days after reaching 90% of RATED POWER). [Reference: NRC Safety Evaluation for License Amendments 270/269, issued October 19, 2010] The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. If reactor power is reduced below 90% of RATED POWER before completion of the RCS flow surveillance, the 7 day period shall be exited, and a separate 7 day period shall be entered when the required condition of reaching 90% of RATED POWER is subsequently achieved. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS Table 4.1-2B

Item 1 - RCS Coolant Liquid Samples

DOSE EQUIVALENT I-131 - This surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The SFCP 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

DOSE EQUIVALENT XE-133 - This surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days per the SFCP. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken or equivalent sampling method. This surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SFCP 7 day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in this calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

reduces the consequences of a steam line break inside the containment by stopping the entry of feedwater.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System decay heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lessons Learned Task Force Status Report," NUREG-0578, item 2.1.7.b.

Loss of Power - 4.16 KV Emergency Bus Negative Sequence Voltage (Open Phase)

The Open Phase Isolation System (OPIS) must be operable when an Emergency bus is required to be operable and is fed from an offsite source. Normally, offsite power is provided to the Emergency buses through the Reserve Station Service Transformers (RSSTs); therefore, the OPIS must be operable any time a required Emergency bus is fed from an RSST. With the generator offline and offsite power provided to an Emergency bus through a Main Step-Up Transformer (i.e., back-feed configuration), the OPIS will detect open phase conditions and initiate a transfer of an Emergency bus for specific open phase conditions. Therefore, the OPIS must be operable any time a required Emergency bus is fed from a Main Step-Up Transformer during backfeed.

OPIS relays are installed on each Emergency bus. The OPIS relays are designed with logic which disables protection when the respective Emergency bus normal feeder breaker is open. Thus, the OPIS protection is passive and is not required to be active when an Emergency bus is not being fed from an offsite source (e.g., being powered by an emergency diesel generator.)

Setting Limits

1. The high containment pressure limit is set at about 8% of design containment pressure. Initiation of safety injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The high-high containment pressure limit is set at about 21% of design containment pressure. Initiation of containment spray and steam line isolation protects against large loss-of-coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure setpoint for safety injection actuation is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.[2] The setting limit (in units of psig) is based on nominal atmospheric pressure.

4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its HOT SHUTDOWN value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[3]

TS action statement 3.16.B.1.a.2 provides an allowance to avoid unnecessary testing of an OPERABLE EDG(s). If it can be determined that the cause of an inoperable EDG does not exist on the OPERABLE EDG(s), operability testing does not have to be performed. If the cause of the inoperability exists on the other EDG(s), then the other EDG(s) would be declared inoperable upon discovery, and the applicable required action(s) would be entered. Once the failure is repaired, the common cause failure no longer exists and the operability testing requirement for the OPERABLE EDG(s) is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG(s), performance of the operability test within 24 hours provides assurance of continued operability of those EDG(s).

In the event the inoperable EDG is restored to OPERABLE status prior to completing the operability testing requirement for the OPERABLE EDG(s), the corrective action program will continue to evaluate the common cause possibility, including the other unit's EDG or the shared EDG. This continued evaluation, however, is no longer under the 24-hour constraint imposed by the action statement.

According to Generic Letter 84-15 (Ref. 6), 24 hours is reasonable to confirm that the OPERABLE EDG(s) is not affected by the same problem as the inoperable EDG.

Reserve Station Service Transformer (RSST) C is the primary offsite power source for the 1H and 2J Emergency Buses via transfer bus F. To facilitate the replacement of RSST-C and associated cabling during the fall 2018 Unit 2 refueling outage, Technical Specification 3.16.B.2 is modified by a footnote permitting the use of a temporary, one time, 21-day allowed outage time (AOT). The 21-day AOT will permit Unit 1 to continue to operate for 21 days. While RSST-C is unavailable during replacement, transfer bus F will be powered from the dependable alternate source (i.e, backfeed through the Unit 2 Main Step-up Transformer/2C Station Services Transformer). The backfeed power supply will allow transfer bus F to perform its normal function while RSST-C is being replaced. Prior to entry into the 21-day AOT, the following actions shall be taken:

1. Within 30 days prior to entering the temporary 21-day AOT, functionality of the Alternate AC (AAC) System (i.e., the supplemental power source) shall be verified.

2. During the 21-day AOT, the functionality of the AAC System shall be checked once per shift. If the AAC System becomes non-functional at any time during the 21-day AOT, it shall be restored to functional status within 24 hours, or the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

The verification of functionality of the AAC System prior to entering the temporary 21-day AOT will be based on the previous satisfactory quarterly test. The once per shift functionality check will be performed during shiftly operator rounds.

In addition to verifying and checking functionality of the AAC System prior to and during the temporary 21-day AOT, the following actions will also be taken:

- Weather conditions will be monitored and preplanned maintenance will not be scheduled if severe weather conditions are anticipated.
- The system load dispatcher will be contacted once per day to ensure no significant grid perturbations (high grid loading unable to withstand a single contingency of line or generation outage) are expected during the temporary 21-day AOT.
- Component testing or maintenance of safety systems and important non-safety equipment in the offsite power systems that can increase the likelihood of a plant transient (unit trip) or LOOP will be avoided. In addition, no discretionary switchyard maintenance will be performed.
- TS required systems, subsystems, trains, components, and devices that depend on the remaining power sources will be verified to be operable and positive measures will be provided to preclude subsequent testing or maintenance activities on these systems, subsystems, trains, components, and devices.
- Operation or maintenance of plant equipment when its redundant equipment or train is out of service will be controlled in accordance with procedure OP-SU-601, "Protected Equipment." The Unit 1 steam-driven Auxiliary Feedwater Pump will be controlled as "Protected Equipment" during the temporary 21-day AOT.
- The status of the AAC diesel generator, EDGS, RSST-A and RSST-B will be monitored once per shift.

References

(1) UFSAR Section 8.5 Emergency Power System

(2) UFSAR Section 9.3 Residual Heat Removal System

(3) UFSAR Section 9.4 Component Cooling System

(4) UFSAR Section 10.3.2 Auxiliary Steam System

(5) UFSAR Section 10.3.5 Condensate and Feedwater System

(6) Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," dated July 2, 1984