VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 20, 2022

10 CFR 50.90

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

Serial No.: 22-035
NRA/GDM: 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
PROPOSED LICENSE AMENDMENT REQUEST
ADMINISTRATIVE CHANGES TO SUBSEQUENT RENEWED OPERATING LICENSES AND TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion Energy Virginia) proposes a change to the Surry Power Station (SPS) Units 1 and 2 Subsequent Renewed Operating Licenses (OLs), DPR-32 and 37, respectively, and the associated Technical Specifications (TS) to delete expired License Conditions (LCs) and TS that were previously included in the Surry Units 1 and 2 OLs and TS to support one-time plant modifications or the initial implementation of plant programs. Administrative changes and correction of editorial errors are also being addressed. The proposed change is administrative in nature and will therefore not result in any changes to plant design or operation. Associated TS Basis changes are also included for the NRC's information. Attachment 1 provides a description and assessment of the proposed change, and Attachments 2 and 3 provide marked-up and typed proposed OLs, TS and Basis pages, respectively.

Dominion Energy Virginia has evaluated the proposed administrative amendment request and has determined 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. Due to the administrative nature of the proposed LAR, 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 change has been reviewed and approved by the Facility Safety Review Committee. Dominion Energy Virginia requests approval of the proposed change by June 30, 2023, with a 30-day implementation period.

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

Respectfully,

Douglas C. Lawrence
Vice President – Nuclear Engineering and Fleet Support

Commitments contained in this letter: None

Attachments:

1. Description and Assessment
2. Proposed Subsequent Renewed Operating Licenses, Technical Specifications, and Basis Pages (Marked-up)
3. Proposed Subsequent Renewed Operating Licenses, Technical Specifications, and Basis Pages (Typed)

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mr. Douglas C. Lawrence, 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 20th day of June, 2022.

My Commission Expires: 12/31/24.

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2024

Notary Public

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Attachment 1

DESCRIPTION AND ASSESSMENT

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

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion Energy Virginia) proposes a change to the Surry Power Station (SPS) Units 1 and 2 Subsequent Renewed Operating Licenses (OLs), DPR-32 and 37, respectively, and the associated Technical Specifications (TS) to delete expired License Conditions (LCs) and TS that were previously included in the Surry Units 1 and 2 OLs and TS to support one-time plant modifications or the initial implementation of plant programs. Administrative changes and correction of editorial errors are also being addressed by the proposed change. The proposed change is administrative in nature and will therefore not result in changes to plant design or operation. Associated TS Basis changes are also included for the NRC's information.

2.0 DETAILED DESCRIPTION OF THE PROPOSED CHANGE

The proposed change to the SPS Units 1 and 2 OLs and accompanying TS: 1) deletes expired, obsolete information and requirements associated with previously completed plant modifications and plant program implementation requirements, and 2) makes administrative changes and corrects minor editorial errors in various sections of the TS. The affected LCs and TS, and their associated license amendments or reference documents, as applicable, are provided in Table 1. A description of, and the reason for, each proposed change is provided in Section 2.

TABLE 1

Proposed Administrative Changes to the SPS Units 1 and 2 Subsequent Renewed Operating Licenses DPR-32 and 37 and Technical Specifications

LC / TS	Reason for Change	Proposed Change	Reference Documents
LC 3.R TS 1.D TS 3.23.C.2.a.1 TS 3.23.C.2.b.1	Completion of Main Control Room (MCR) and Emergency Switchgear Room (ESGR) Chilled Water Piping Replacement Project	Delete expired LC 3.R Delete expired footnotes associated with: TS 1.D TS 3.23.C.2.a.1 TS 3.23.C.2.b.1	Amendments 258/257

TABLE 1

Proposed Administrative Changes to the SPS Units 1 and 2 Subsequent Renewed Operating Licenses DPR-32 and 37 and Technical Specifications

LC / TS	Reason for Change	Proposed Change	Reference Documents
LC 3.S	Completion of first performance of MCR/ESGR Envelope Habitability Program Testing, Assessment and Measurement	Delete expired LC 3.S	Amendments 260/260
LC 3.T	Measurement Uncertainty Recapture (MUR) Power Uprate License Condition 3.T commitments have been completed	Delete expired LC 3.T	Amendments 269/268
TS 3.1 Pages: 3.1-13 3.1-14a 3.1-14b	Incorrect Unit 2 Amendment No. 250 included on TS Pages	Correct Unit 2 License Amendment No. on bottom of affected associated TS pages	Amendments 267/266
TS 3.10	Incorrect reference to TS 3.23.C in TS 3.10.A.13 and 14 and TS 3.10.B.6 and 7	Correct referenced TS section numbers	Amendments 260/260
TS 3.16	Completion of Unit 2 Reserve Station Service Transformer (RSST) C and Associated 5-KV Cable Replacement	Delete expired footnotes and Basis discussion associated with completed modifications	Amendments 293/293 and 297/297
TS 4.9.A	Incorrect reference to TS Section 3.7.E	Correct referenced TS section to TS 3.7.D	Amendments 228/228
TS 6.1.2.2.d	Operations organizational position title change	Revise position title	N/A
TS 6.7.B	Typographical error	Correct typo	NRC Environmental Qualification Order dated October 24, 1980

2.1 Deletion of Expired License Conditions, Technical Specifications, and Associated Basis Text

2.1.1 LC 3.R, TS 1.D OPERABLE Definition Footnote, and TS 3.23.A.2.a.1 and b.1 Footnotes - Replacement of MCR/ESGR Air-Conditioning System (ACS) Chilled Water Piping

SPS Units 1 and 2 Amendments 258 and 257 (Reference 6.1) implemented a temporary LC and TS to facilitate the replacement of the MCR/ESGR ACS chilled water piping. Specifically, these amendments added LC 3.R to each unit's Operating License (OL) and a footnote to TS 3.23.C.2.a.1 and 3.23.C.2.b.1 permitting the use of temporary 45-day and 14-day allowed outage time (AOT) extensions four times in a 24-month time span to permit replacement of degraded MCR/ESGR ACS chilled water piping. The amendments also revised the definition of OPERABLE in TS 1.D specific to the MCR/ESGR ACS air handling units on the operating chilled water loop by including a footnote to permit either the normal or emergency electrical power source to be capable of performing its related support function for the purpose of performing TS-required surveillances that would render an emergency diesel generator inoperable. The chilled water piping replacement work was completed in 2008, and the associated design change package was closed out in June 2009. Furthermore, the 24-month time span permitted for use of the four AOT extensions began when the first AOT extension was entered on February 4, 2008, and therefore expired on February 4, 2010. Consequently, LC 3.R and the footnotes associated with TS 1.D, 3.23.A.2.a.1 and 3.23.A.2.b.1 have expired and are therefore being deleted. (It should be noted that the TS 3.23 footnote refers to TS 3.23.C.2.a.1 and b.1. TS 3.23.C was subsequently renumbered as TS 3.23.A by Amendments 260/260 as noted in Item 2.2.1 below.) Related text in the TS 3.23 Basis is also being deleted as it refers to the work completed under the expired temporary TS AOTs.

2.1.2 LC 3.S – Control Room Envelope (CRE) Habitability Program

SPS Units 1 and 2 Amendments 260/260 (Reference 6.2) incorporated OL and TS changes pursuant to Technical Specification Task Force (TSTF) Traveler TSTF-448, Revision 3, "Control Room Habitability." One of the changes was the incorporation of LC 3.S into the OLs which specified schedular requirements associated with: 1) the first performance of Surveillance Requirement (SR) 4.18, 2) the periodic assessment of MCR/ESGR envelope habitability specified in TS 6.4.R.3.b, and 3) the periodic measurement of MCR/ESGR envelope pressure as specified in TS 6.4.R.4. The first performance of these three requirements associated with CRE habitability surveillance, assessment, and measurement were completed as indicated in Table 2.1.

Since the first performance of the TS 4.18 surveillance, TS 6.4.R.3.b periodic assessment, and TS 6.4.R.4 periodic measurement of the CRE have been completed in accordance with the LC 3.S schedular requirements, LC 3.S is no longer required and is being deleted.

TABLE 2.1 - COMPLETION OF LICENSE CONDITION 3.S(1), (2), AND (3) *(Control Room Envelope Habitability Requirements)*		
LC #	**LC Due Date**	**Completion Date**
3.S(1)	07/18/11	10/15/10
3.S(2)	06/24/09	06/18/09
3.S(3)	12/04/08	12/01/08

2.1.3 LC 3.T – Measurement Uncertainty Recapture (MUR)

SPS Units 1 and 2 Amendments 269/268 (Reference 6.3) implemented an MUR power uprate from 2546 to 2587 MWt. The license amendments included new LC 3.T that required the completion of sixteen commitments, fourteen of which were required to be completed prior to exceeding 2546 MWt as noted below.

TABLE 2.2 – MUR License Condition Commitments and Due Dates	
COMMITMENT	**SCHEDULED COMPLETION DATE**
1. VEPCO will perform the final acceptance of the Surry 1/2 uncertainty analysis to ensure the results are bounded by the statements contained in the LAR	Prior to operating above 2546 MWt (98.4% RP)
2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls	Prior to operating above 2546 MWt (98.4% RP)
3. Revise procedures, programs, and documents for the new UFM (including transducer replacement)	Prior to operating above 2546 MWt (98.4% RP)
4. Appropriate personnel will receive training on the UFM and affected procedures	Prior to operating above 2546 MWt (98.4% RP)
5. The FAC CHECWORKS SFA models will be updated to reflect the MUR PU conditions	Prior to operating above 2546 MWt (98.4% RP)
6. Simulator changes and validation will be completed	Prior to operating above 2546 MWt (98.4% RP)
7. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels	Prior to operating above 2546 MWt (98.4% RP)

TABLE 2.2 – MUR License Condition Commitments and Due Dates	
8. Process UFSAR changes in accordance with 10 CFR 50.59	In accordance with 10 CFR 50.71(e)
9. UFM commissioning and calibration will be completed	Prior to operating above 2546 MWt (98.4% RP)
10. Confirm flow normalization factors	Prior to operating above 2546 MWt (98.4% RP)
11. Rescaling and calibration of main turbine first stage pressure input to AMSAC	Prior to operating above 2546 MWt (98.4% RP)
12. Determine EQ service life for excore detectors	Prior to operating above 2546 MWt (98.4% RP)
13. The excore neutron detectors are scheduled to be replaced	Unit 1: fall 2010 Refueling Outage Unit 2: spring 2011 Refueling Outage
14. Revise EOP setpoints	Prior to operating above 2546 MWt (98.4% RP)
15. The UFM feedwater flow and temperature data will be compared to the feedwater flow venturis output and the feedwater RTD output	Prior to operating above 2546 MWt (98.4% RP)
16. For the applicable UFSAR Chapter 14 events, Surry 1/2 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A.	Prior to operating above 2546 MWt (98.4% RP)

As required by LC 3.T, Commitments 1 through 7, 9 through 12, and 14 through 16 were completed for each unit prior to exceeding 2546 MWt, which occurred on December 8, 2010, for SPS Unit 1 and on June 30, 2011, for SPS Unit 2.

The remaining two items, Commitments 8 and 13, required the UFSAR to be updated in accordance with 10 CFR 50.71(e), and the Unit 1 and Unit 2 excore neutron detectors to be replaced by the completion of the fall 2010 and spring 2011 refueling outages (RFOs), respectively. Commitment 8 was completed when the SPS Units 1 and 2 UFSAR was updated in Revision 43 dated September 27, 2011 (Serial No. 11-459) [ADAMS Accession No. ML11287A165] to reflect the MUR power uprate. Commitment 13 was met upon the replacement of the SPS Units 1 and 2 excore neutron detectors on November 30, 2010, during the Unit 1 fall 2010 RFO and on May 27, 2011, during the Unit 2 spring 2011 RFO, respectively. Consequently, LC 3.T is no longer required and is being deleted.

2.1.4 TS 3.16, "Emergency Power System," - Temporary Extended Allowed Outage Times (AOTs) for Replacement of RSST C and its associated 5-KV Cable

SPS Units 1 and 2 Amendments 293/293 (Reference 6.4) implemented a temporary, one-time, 21-day AOT in a footnote to TS 3.16, Action B.2, to permit replacement of the Unit 2 RSST C. Amendments 297/297 (Reference 6.5) revised the footnote included in Amendments 293/293 to permit a temporary, one-time, 14-day AOT for replacement of the 5-KV cabling associated with Unit 2 RSST C consistent with the specified footnote conditions. RSST C was replaced during the fall 2018 Unit 2 RFO, and its associated cabling was replaced during the spring 2020 Unit 2 RFO. The one-time, 14-day AOT was exited on May 24, 2020. Consequently, the temporary, one-time AOT footnote to TS 3.16, Action B.2, has expired and can be deleted. Related text in the TS 3.16 Basis is also being deleted as it discusses the alternate dependable power source that would be used while RSST C was out of service, as well as the additional actions to be taken prior to entering the temporary, one-time AOT permitted by the footnote.

2.2 Administrative Changes and Corrections

2.2.1 TS 3.10, "Refueling"

As noted above, SPS Units 1 and 2 Amendments 260/260 (Reference 6.2) incorporated TS changes pursuant to TSTF-448, Revision 3, "Control Room Habitability." The amendments included a revision to then TS 3.23, "Main Control Room and Emergency Switchgear Room Ventilation and Air Conditioning System." The amendments separated TS 3.23 into two separate TS sections: TS 3.21, "Main Control Room and Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)," and TS 3.23, "Main Control Room and Emergency Switchgear Room Air Conditioning System." As part of that change, existing TS 3.23.A and B, which addressed the MCR/ESGR EVS, were revised and relocated to new TS section 3.21, while TS 3.23.C, which only addressed the MCR/ESGR Air Conditioning System (ACS) was renumbered as TS 3.23.A.

However, it was not recognized at the time those changes were made that TS 3.10 included references to TS 3.23.C, which had been subsequently renumbered as TS 3.23.A by Amendments 260/260. Consequently, the proposed change revises TS 3.10.A.13 and 14 and TS 3.10.B.6 and 7 to reference TS 3.23.A instead of TS 3.23.C, which no longer exists.

2.2.2 TS 4.9, "Radioactive Gas Storage Monitoring System"

SPS Units 1 and 2 Amendments 228/228 (Reference 6.6) implemented changes to the Reactor Protection System (RPS) and the Engineered Safety Features Actuation System (ESFAS) instrumentation systems. The amendments revised the surveillance test intervals, AOTs, bypass times, and limiting conditions for operation (LCO) for those systems. As part of those changes, TS 3.7.A was deleted and the remaining sections were renumbered. This change resulted in TS 3.7.E, which was associated with the explosive gas monitoring instrumentation channel, being renumbered as TS 3.7.D. However, it was not identified at the time the change was made that TS 3.7.E was referenced in TS 4.9, "Radioactive Gas Storage Monitoring System." Consequently, the reference to TS 3.7.E in TS 4.9.A was not changed to reference the revised TS number 3.7.D. Therefore, the proposed change revises TS 4.9.A to reference TS 3.7.D rather than TS 3.7.E.

2.2.3 TS 6.1.2, "Organization"

TS 6.1.2 in the Administrative Controls section of the TS includes the Unit Staff requirements in TS 6.1.2.2. TS 6.1.2.2.d currently states, "The Supervisor Nuclear Shift Operations shall hold an active Senior Reactor Operator License for Surry Power Station." The title "Supervisor Nuclear Shift Operations" was subsequently changed to "Superintendent Nuclear Shift Operations." The proposed change revises TS 6.1.2.2.d to reflect the current position title. The position responsibilities and qualification requirements are unchanged.

2.2.4 TS 6.7, "Environmental Qualification"

TS 6.7 was incorporated into the SPS Units 1 and 2 TS pursuant to an NRC Order dated October 24, 1980, "Orders for Modification of Licenses Concerning Environmental Qualification of Safety-Related Electrical Equipment" (Reference 6.7). When TS 6.7.B was included in the TS as required by the Order, an editorial error was inadvertently introduced. Specifically, the first line of TS 6.7.B includes the words, "By **on** later than..." as opposed to the correct wording, "By **no** later than..." The proposed change corrects the TS wording discrepancy.

3.0 TECHNICAL EVALUATION

The proposed change to the SPS Units 1 and 2 Subsequent Renewed OLs and TS is administrative in nature and does not affect how plant equipment is operated or maintained. The change deletes expired LCs and TS footnotes associated with completed plant modifications and programmatic implementation requirements, makes administrative changes to correct errors introduced in previous license amendments, and updates a personnel title. No changes to the physical plant or analytical methods are described, and there are no impacts on the Updated Final Safety Analysis Report (UFSAR) accident analyses.

4.0 REGULATORY EVALUATION

4.1 Determination of No Significant Hazards Consideration

Virginia Electric and Power Company (Dominion Energy Virginia) has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed administrative change to the Surry Power Station (SPS) Units 1 and 2 Subsequent Renewed Operating Licenses (SROLs) DPR-32 and DPR-37 and the Technical Specifications (TS). The proposed change deletes expired License Conditions (LCs) and TS footnotes associated with completed plant modifications and programmatic implementation requirements, makes administrative changes to correct minor editorial errors introduced in previous license amendments, and updates a personnel title. In accordance with the criteria set forth in 10 CFR 50.92, Dominion Energy Virginia has performed an analysis of the proposed change and concluded that it does not represent a significant hazards consideration. The following discussion is provided in support of this conclusion:

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

   Response: No

   The proposed change to delete expired, temporary LCs and TS, correct minor TS administrative errors and update a personnel title does not impact the condition or performance of any plant structure, system, or component. The proposed administrative change does not affect the initiators of any previously analyzed event or the assumed mitigation of accident or transient events. As a result, the proposed change to the SPS Units 1 and 2 SROLs and TS does not involve an increase in the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by the proposed administrative change.

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

   Response: No

   The proposed change to delete expired, temporary LCs and TS, correct minor TS editorial errors and update a personnel title is administrative in nature, and therefore does not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions, and no new failure modes are being introduced. Therefore, the proposed administrative change to the Surry SROLs and

TS does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

   Response: No

   The proposed change to delete expired, temporary LCs and TS, correct minor TS editorial errors and update a personnel title is administrative in nature and therefore does not affect the acceptance criteria for any analyzed event, nor is there a change to any safety limit. The proposed change does not affect any structures, systems or components or their capability to perform their intended functions and does not alter the way safety limits, limiting safety system settings or limiting conditions for operation are determined. Neither the safety analyses nor the safety analyses acceptance criteria are affected by the proposed change. Therefore, the proposed administrative change to the Surry SROLs and TS does not involve a reduction in a margin of safety.

4.2 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.3 Precedents

Numerous licensees have submitted LARs to make administrative and editorial changes including the deletion of expired, obsolete and/or completed requirements contained in their OLs and TS. A sample of these LARs is provided below:

- By letter dated March 31, 2021, DTE Electric Company submitted an LAR for Fermi 2 Power Plant to revise its TS to remove obsolete information, make minor corrections, and make miscellaneous editorial changes. (ADAMS Accession No. ML21090A194)

- By letter dated June 12, 2018, Energy Northwest submitted an LAR for Columbia Generating Station to clean-up its OL and TS. (ADAMS Accession No. ML18163A351)

- By letter dated February 23, 2017, Exelon Generation Company, LLC, submitted an administrative LAR for the Braidwood and Byron Power Stations' Units 1 and 2 to remove time, cycle, or modification-related items from their respective OLs and TS and to make editorial and formatting changes. (ADAMS Accession No. ML17055A631)

- By letter dated February 12, 2016, NextEra Energy Point Beach, LLC, submitted an administrative LAR for Point Beach Nuclear Plant that removed LCs that had been completed and were no longer in effect. (ADAMS Accession No. ML16043A217)
- By letter dated July 11, 2014, Exelon Generation Company, LLC, submitted an administrative LAR for Peach Bottom Atomic Power Station, Units 2 and 3, which deleted an LC associated with a one-time commitment. (ADAMS Accession No. ML14192B143)

5.0 ENVIRONMENTAL ASSESSMENT

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described above, the proposed administrative change does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed administrative change to the SPS Units 1 and 2 SROLs and TS does not involve the installation of any new equipment or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Plant operation is not affected in any manner by this proposed administrative change. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed administrative change does not involve plant physical changes or changes in the method of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above assessment, Dominion Energy Virginia concludes that the proposed administrative change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment or impact statement by the Commission.

6.0 REFERENCES

6.1 Letter from US NRC to Virginia Electric and Power Company dated January 23, 2008, "Subject: Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Replacement of Main Control Room and Emergency Switchgear Room Air-Conditioning System Chilled Water Piping (TAC Nos. MD4622 and MD4623)," [ADAMS Accession No. ML073480287].

6.2 Letter from US NRC to Virginia Electric and Power Company dated July 7, 2008, "Subject: Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Control Room Habitability (TAC Nos. MD6139 and MD6140)," [ADAMS Accession No. ML081750690].

6.3 Letter from US NRC to Virginia Electric and Power Company dated September 24, 2010, "Subject: Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME3293 and ME3294)," [ADAMS Accession No. ML101750002].

6.4 Letter from US NRC to Virginia Electric and Power Company dated October 5, 2018, "Subject: Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Revising Technical Specifications Section 3.16, 'Emergency Power System,' for a Temporary 21-Day Allowed Outage Time (EPID L-2017-LLA-0380)," [ADAMS Accession No. ML18261A099].

6.5 Letter from US NRC to Virginia Electric and Power Company dated March 16, 2020, "Subject: Surry Power Station, Units 1 and 2, Issuance of Amendments Nos. 297 and 297, to Revise Technical Specifications 3.16, 'Emergency Power System,' to Allow a One-Time 14-Day Allowed Outage Time for Replacement of 5-KV Cables Associated with Reserve Station Service Transformer C (EPID No. L-2019-LLA-0244)," [ADAMS Accession No. ML20058F966].

6.6 Letter from US NRC to Virginia Electric and Power Company dated August 31, 2001, "Subject: Surry Units 1 and 2 - Issuance of Amendments Re: Changes to Surveillance Test Intervals and Allowed Outage Times for Instrumentation Systems (TAC Nos. MA9355 And MA9356)," [ADAMS Accession No. ML012480506].

6.7 Letter from US NRC to Virginia Electric and Power Company dated October 24, 1980, "Orders for Modification of Licenses Concerning Environmental Qualification of Safety-Related Electrical Equipment," [ADAMS Accession No. ML012490039].

Attachment 2

PROPOSED SUBSEQUENT RENEWED OPERATING LICENSES, TECHNICAL SPECIFICATIONS, AND BASIS PAGES (MARKED-UP)

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

(3) Actions to minimize release to include consideration of:

Deleted by Amendment XXX

a. Water spray scrubbing
b. Dose to onsite responders

R. ~~As discussed in the footnote to Technical Specifications 3.23.C.2.a.1 and 3.23.C.2.b.1, the use of temporary 45-day and 14-day allowed outage time extensions to permit replacement of the Main Control Room and Emergency Switchgear Room Air Conditioning System chilled water piping shall be in accordance with the basis, risk evaluation, equipment unavailability restrictions, and compensatory actions provided in the licensee's submittal dated February 26, 2007 (Serial No. 07-0109) and in the associated supplemental transmittals, as approved by the NRC Safety Evaluation.~~

Deleted by Amendment XXX

S. ~~Upon implementation of Amendment No. 260 adopting TSTF-448, Revision 3, the determination of Main Control Room/Emergency Switchgear Room (MCR/ESGR) envelope unfiltered air inleakage as required by TS SR 4.18 in accordance with TS 6.4.R.3.a, the assessment of MCR/ESGR envelope habitability as required by Specification 6.4.R.3.b, and the measurement of MCR/ESGR envelope pressure as required by Specification 6.4.R.4, shall be considered met. Following implementation:~~

~~(1) The first performance of SR 4.18, in accordance with Specification 6.4.R.3.a, shall be within the specified frequency of 6 years plus the 18-month allowance of SR 4.0.2, as measured from January 18, 2004, the date of the most recent successful tracer gas test, as stated in the April 22, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~

~~(2) The first performance of the periodic assessment of MCR/ESGR envelope habitability, Specification 6.4.R.3.b, shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from January 18, 2004, the date of the most recent successful tracer gas test, as stated in the April 22, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~

~~(3) The first performance of the periodic measurement of MCR/ESGR envelope pressure, Specification 6.4.R.4, shall be within 18 months, plus the 138 days allowed by SR 4.0.2, as measured from January 19, 2007, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.~~

T. Deleted by Amendment XXX

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~1. VEPCO will perform the final acceptance of Surry 1 uncertainty analysis to ensure the results are bounded by the statements contained in the LAR (Attachment 5, Section I.1.D.4.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~3. Revise procedures, programs, and documents for the new UFM (including transducer replacement) (Attachment 5, Sections I.1, I.1.D.1, I.1.H, VII.1, VII.2.A, and VII.4).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5, Sections I.1.D.1.1, VII.2.A, VII.2.D, and VII.3).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~5. The FAC CHECWORKS SFA models will be updated to reflect the MUR PU conditions (Attachment 5, Section IV.1.E.iii).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~6. Simulator changes and validation will be completed (Attachment 4, Section VII.2.C).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~7. Revise existing plant operating procedures related to temporary operation above full steady state licenses power levels (Attachment 5, Section VII.4).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~8. Process UFSAR changes in accordance with 10 CFR 50.59 (Attachment 1, Section 3.0).~~	~~In accordance with 10 CFR 50.71(e)~~

~~T. (Continued)~~

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~9. UFM commissioning and calibration will be completed (Attachment 5, Section I.1.D.2.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~10. Confirm flow normalization factors (Attachment 5, Section I.1.G).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~11. Rescaling and calibration of main turbine first stage pressure input to AMSAC (Attachment 5, Sections II.2.28, VII.2.B, VIII.2, and VIII.3).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~12. Determine EQ service life for excore detectors (Attachment 5, Sections III.2.A and V.1.C).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~13. The excore neutron detectors are scheduled to be replaced (Attachment 5, Section V.1.C).~~	~~Unit 1: fall 2010 Refueling Outage.~~ ~~Unit 2: spring 2011 Refueling Outage.~~
~~14. Revise EOP setpoints (Attachment 5 Section VII.2.A).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~15. The UFM feedwater flow and temperature data will be compared to the feedwater flow venturis output and the feedwater RTD output (Attachment 5, Section I.1.D.2.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~

~~T. (Continued)~~

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~16. For the applicable UFSAR Chapter 14 events, Surry 1 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A.~~ ~~If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 1 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A:~~ ~~• Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only);~~ ~~• Section 14.2.8 - Excessive Load Increase Incident;~~ ~~• Section 14.2.9 - Loss of Reactor Coolant Flow; and~~ ~~• Section 14.2.10 - Loss of External Electrical Load~~	~~Prior to operating above 2546 MWt (98.4% RP).~~

U. Deleted by Amendment No. 289

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020.

Deleted by Amendment XXX

(3) Actions to minimize release to include consideration of:

a. Water spray scrubbing
b. Dose to onsite responders

Deleted by Amendment XXX

R. ~~As discussed in the footnote to Technical Specifications 3.23.C.2.a.1 and 3.23.C.2.b.1, the use of temporary 45-day and 14-day allowed outage time extensions to permit replacement of the Main Control Room and Emergency Switchgear Room Air Conditioning System chilled water piping shall be in accordance with the basis, risk evaluation, equipment unavailability restrictions, and compensatory actions provided in the licensee's submittal dated February 26, 2007 (Serial No. 07-0109) and in the associated supplemental transmittals, as approved by the NRC Safety Evaluation.~~

S. ~~Upon implementation of Amendment No. 260 adopting TSTF-448, Revision 3, the determination of Main Control Room/Emergency Switchgear Room (MCR/ESGR) envelope unfiltered air inleakage as required by TS SR 4.18 in accordance with TS 6.4.R.3.a, the assessment of MCR/ESGR envelope habitability as required by Specification 6.4.R.3.b, and the measurement of MCR/ESGR envelope pressure as required by Specification 6.4.R.4, shall be considered met. Following implementation:~~

~~(1) The first performance of SR 4.18, in accordance with Specification 6.4.R.3.a, shall be within the specified frequency of 6 years plus the 18-month allowance of SR 4.0.2, as measured from January 18, 2004, the date of the most recent successful tracer gas test, as stated in the April 22, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~

~~(2) The first performance of the periodic assessment of MCR/ESGR envelope habitability, Specification 6.4.R.3.b, shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from January 18, 2004, the date of the most recent successful tracer gas test, as stated in the April 22, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~

~~(3) The first performance of the periodic measurement of MCR/ESGR envelope pressure, Specification 6.4.R.4, shall be within 18 months, plus the 138 days allowed by SR 4.0.2, as measured from January 19, 2007, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.~~

T. Deleted by Amendment XXX

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~1. VEPCO will perform the final acceptance of Surry 2 uncertainty analysis to ensure the results are bounded by the statements contained in the LAR (Attachment 5, Section I.1.D.4.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~3. Revise procedures, programs, and documents for the new UFM (including transducer replacement) (Attachment 5, Sections I.1, I.1.D.1, I.1.H, VII.1, VII.2.A, and VII.4).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5, Sections I.1.D.1.1, VII.2.A, VII.2.D, and VII.3).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~5. The FAC CHECWORKS SFA models will be updated to reflect the MUR PU conditions (Attachment 5, Section IV.1.E.iii).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~6. Simulator changes and validation will be completed (Attachment 4, Section VII.2.C).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~7. Revise existing plant operating procedures related to temporary operation above full steady-state licenses power levels (Attachment 5, Section VII.4).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~8. Process UFSAR changes in accordance with 10 CFR 50.59 (Attachment 1, Section 3.0).~~	~~In accordance with 10 CFR 50.71(e)~~

T. ~~(Continued)~~

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~9. UFM commissioning and calibration will be completed (Attachment 5, Section I.1.D.2.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~10. Confirm flow normalization factors (Attachment 5, Section I.1.G).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~11. Rescaling and calibration of main turbine first stage pressure input to AMSAC (Attachment 5, Sections II.2.28, VII.2.B, VIII.2, and VIII.3).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~12. Determine EQ service life for excore detectors (Attachment 5, Sections III.2.A and V.1.C).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~13. The excore neutron detectors are scheduled to be replaced (Attachment 5, Section V.1.C).~~	~~Unit 1: fall 2010 Refueling Outage.~~ ~~Unit 2: spring 2011 Refueling Outage.~~
~~14. Revise EOP setpoints (Attachment 5 Section VII.2.A).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~
~~15. The UFM feedwater flow and temperature data will be compared to the feedwater flow venturis output and the feedwater RTD output (Attachment 5, Section I.1.D.2.1).~~	~~Prior to operating above 2546 MWt (98.4% RP).~~

~~T. (Continued)~~

~~COMMITMENT~~	~~SCHEDULED COMPLETION DATE~~
~~16. For the applicable UFSAR Chapter 14 events, Surry 2 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A.~~ ~~If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 2 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A:~~ ~~• Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only);~~ ~~• Section 14.2.8 - Excessive Load Increase Incident;~~ ~~• Section 14.2.9 - Loss of Reactor Coolant Flow; and~~ ~~• Section 14.2.10 - Loss of External Electrical Load~~	~~Prior to operating above 2546 MWt (98.4% RP).~~

U. Deleted by Amendment 289

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

5. REACTOR CRITICAL

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

6. POWER OPERATION

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

7. REFUELING OPERATION

Any operation involving movement of core components when the vessel head is unbolted or removed.

D. OPERABLE

A system, subsystem, train, component, or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and~~*~~ emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). The system or component shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

E. PROTECTIVE INSTRUMENTATION LOGIC

1. ANALOG CHANNEL

An arrangement of components and modules as required to generate a single protective action digital signal when required by a unit condition. An analog channel loses its identity when single action signals are combined.

~~* For the purpose of performing Technical Specification-required surveillances that render an emergency diesel generator inoperable, the definition of OPERABLE for the air handling units on the operating chilled water loop is modified to require the normal or emergency electrical power source to be capable of performing its related support function. This footnote shall only apply to TS 3.23.C.2.a.1, TS 3.23.C.2.b.1, and TS 3.23.C.2.c.1 during the 45-day allowed outage time extensions to permit replacement of the Main Control Room and Emergency Switchgear Room Air Conditioning System chilled water piping.~~

C. RCS Operational LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. RCS operational LEAKAGE shall be limited to:

   a. No pressure boundary LEAKAGE,

   b. 1 gpm unidentified LEAKAGE,

   c. 10 gpm identified LEAKAGE, and

   d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

2.a. If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.

b. If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3. If RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within the limit specified in 3.1.C.1.d, the unit shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

10. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

    This restriction does not apply to the movement of the transfer canal door.

11. Two Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) trains shall be OPERABLE.

    a. With one required train inoperable for reasons other than an inoperable MCR/ESGR envelope boundary, restore the inoperable train to OPERABLE status within 7 days. If the inoperable train is not returned to OPERABLE status within 7 days, comply with Specification 3.10.C.

    b. If two required trains are inoperable or one or more required trains are inoperable due to an inoperable MCR/ESGR envelope boundary, comply with Specification 3.10.C.

12. Manual actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation shall be OPERABLE as specified in TS 3.7.F.

13. Three chillers shall be OPERABLE in accordance with the power supply requirements of Specification 3.23.~~C~~A. With one of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.~~C~~A.1, return the inoperable chiller to OPERABLE status within 7 days or comply with Specification 3.10.C. With two of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.~~C~~A.1, comply with Specification 3.10.C.

14. Eight air handling units (AHUs) shall be OPERABLE in accordance with the operability requirements of Specification 3.23.~~C~~A. With two AHUs inoperable on the shutdown unit, ensure that one AHU is OPERABLE in each unit's main control room and emergency switchgear room, and restore an inoperable AHU to OPERABLE status within 7 days, or comply with Specification 3.10.C. With more than two AHUs inoperable, comply with Specification 3.10.C.

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

   This restriction does not apply to the movement of the transfer canal door.

3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two MCR/ESGR EVS trains shall be OPERABLE.

   a. With one required train inoperable for reasons other than an inoperable MCR/ESGR envelope boundary, restore the inoperable train to OPERABLE status within 7 days. If the inoperable train is not returned to OPERABLE status within 7 days, comply with Specification 3.10.C.

   b. If two required trains are inoperable or one or more required trains are inoperable due to an inoperable MCR/ESGR envelope boundary, comply with Specification 3.10.C.

5. Manual actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation shall be OPERABLE as specified in TS 3.7.F.

6. Three chillers shall be OPERABLE in accordance with the power supply requirements of Specification 3.23.~~C~~. With one of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.~~C~~.1, return the inoperable chiller to OPERABLE status within 7 days or comply with Specification 3.10.C. With two of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.~~C~~.1, comply with Specification 3.10.C.

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7. Eight air handling units (AHUs) shall be OPERABLE in accordance with the operability requirements of Specification 3.23.~~C~~. With two AHUs inoperable on either unit, ensure that one AHU is OPERABLE in each unit's main control room and emergency switchgear room, and restore an inoperable AHU to OPERABLE status within 7 days, or comply with Specification 3.10.C. With more than two AHUs inoperable on a unit, comply with Specification 3.10.C.

C. If any one of the specified limiting conditions for refueling is not met, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease and irradiated fuel shall be placed in a safe position, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

D. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

E. The requirements of 3.0.1 are not applicable.

<u>Basis</u>

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Potential escape paths for fission product radioactivity within containment are required to be closed or capable of closure to prevent the release to the environment. However, since there is no potential for significant containment pressurization during refueling, the Appendix J leakage criteria and tests are not applicable.

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch must be capable of being closed.

2. If a primary source is not available, the unit may be operated for seven (7) days provided the dependable alternate source can be OPERABLE within 8 hours. If specification A-4 is not satisfied within seven (7) days, the unit shall be brought to COLD SHUTDOWN.~~(*)~~

3. One battery may be inoperable for 24 hours provided the other battery and battery chargers remain OPERABLE with one battery charger carrying the DC load of the failed battery's supply system. If the battery is not returned to OPERABLE status within the 24 hour period, the reactor shall be placed in HOT SHUTDOWN. If the battery is not restored to OPERABLE status within an additional 48 hours, the reactor shall be placed in COLD SHUTDOWN.

4. One buried fuel oil storage tank may be inoperable for 7 days for tank inspection and related repair, provided the following actions are taken:

   a. prior to removing the tank from service, verify that 50,000 gallons of replacement fuel oil is available offsite and transportation is available to deliver that volume of fuel oil within 48 hours, and

   b. prior to removing the tank from service and at least once every 12 hours, verify that the remaining buried fuel oil storage tank contains ≥ 17,500 gallons, and

   c. prior to removing the tank from service and at least once every 12 hours, verify that the above ground fuel oil storage tank contains ≥ 50,000 gallons.

~~(*) To facilitate the replacement of the Reserve Station Service Transformer C 5 KV cables to transformer bus F during the Spring 2020 Unit 2 refueling outage, the use of a temporary, one-time, 14-day allowed outage time (AOT) is permitted for the unavailability of a primary source. Prior to entry into and during the 14-day AOT, the following actions shall be taken:~~

- ~~Within 30 days prior to entering the temporary 14-day AOT, functionality of the Alternate AC (AAC) System (i.e., the supplemental power source) shall be verified.~~
- ~~During the 14-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 14-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.~~

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 the RSST C 5KV cables to transformer bus F during the Spring 2020 Unit 2 refueling outage, Technical Specification 3.16.B.2 is modified by a footnote permitting the use of a temporary, one time, 14-day allowed outage time (AOT). The 14-day AOT will permit Unit 1 to continue to operate for 14 days. While RSST C is unavailable to facilitate replacement of the RSST C 5KV cables, transfer bus F will be powered from the dependable alternate source (i.e, backfeed through the Unit 2 Main Step-up Transformer/ Station Service Transformer 2C). The backfeed power supply will allow transfer bus F to perform its normal function while the RSST C 5KV cables are being replaced. Prior to entry into the 14-day AOT, the following actions shall be taken:~~

~~1. Within 30 days prior to entering the temporary 14-day AOT, functionality of the Alternate AC (AAC) System (i.e., the supplemental power source) shall be verified.~~

~~2. During the 14-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 14-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 14-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 14-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 14-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 14-day AOT.~~
- ~~The status of the AAC diesel generator, EDGs, RSST A and RSST B will be monitored once per shift.~~

2. Air Handling Units (AHUs)

   a. Unit 1 air handling units, 1-VS-AC-1, 1-VS-AC-2, 1-VS-AC-6, and 1-VS-AC-7, must be OPERABLE whenever Unit 1 is above COLD SHUTDOWN.

      1. If either any single Unit 1 AHU or two Unit 1 AHUs on the same chilled water loop (1-VS-AC-1 and 1-VS-AC-7 or 1-VS-AC-2 and 1-VS-AC-6)~~*~~ become inoperable, restore operability of the one inoperable AHU or two inoperable AHUs within seven (7) days or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      2. If two Unit 1 AHUs on different chilled water loops and in different air conditioning zones (1-VS-AC-1 and 1-VS-AC-6 or 1-VS-AC-2 and 1-VS-AC-7) become inoperable, restore operability of the two inoperable AHUs within seven (7) days or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      3. If two Unit 1 AHUs in the same air conditioning zone (1-VS-AC-1 and 1-VS-AC-2 or 1-VS-AC-6 and 1-VS-AC-7) become inoperable, restore operability of at least one Unit 1 AHU in each air conditioning zone (1-VS-AC-1 or 1-VS-AC-2 and 1-VS-AC-6 or 1-VS-AC-7) within one (1) hour or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      4. If more than two Unit 1 AHUs become inoperable, restore operability of at least one Unit 1 AHU in each air conditioning zone (1-VS-AC-1 or 1-VS-AC-2 and 1-VS-AC-6 or 1-VS-AC-7) within one (1) hour or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

   b. Unit 2 air handling units, 2-VS-AC-8, 2-VS-AC-9, 2-VS-AC-6, and 2-VS-AC-7 must be OPERABLE whenever Unit 2 is above COLD SHUTDOWN.

      1. If either any single Unit 2 AHU or two Unit 2 AHUs on the same chilled water loop (2-VS-AC-7 and 2-VS-AC-9 or 2-VS-AC-6 and 2-VS-AC-8)~~*~~ become inoperable, restore operability of the one inoperable AHU or two inoperable AHUs within seven (7) days or bring Unit 2 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

~~* The requirements of TS 3.23.C.2.a.1 and TS 3.23.C.2.b.1 may be temporarily suspended according to the limitations noted in items 1, 2, 3 and 4 below for the purpose of replacement of the Main Control Room (MCR) and Emergency Switchgear Room (ESGR) Air Conditioning System (ACS) chilled water piping. The allowed outage time extensions specified in items 1, 2, 3 and 4 of this footnote shall not exceed 24 months beginning with entry into the first temporary extension. Each extension shall be limited to a one-time use which ends when the affected MCR and ESGR ACS components are returned to OPERABLE status. Concurrent use of more than one allowed outage time extension is not permitted.~~

~~1) The time period to accommodate replacement of the chilled water loop C piping in the MCR and the ESGR shall not exceed 45 days.~~

~~2) The time period to accommodate replacement of the chilled water loop A piping in the MCR and the ESGR shall not exceed 45 days.~~

~~3) The time period to accommodate replacement of the chilled water piping in the Mechanical Equipment Room #3 (MER-3) associated with chiller 1-VS-E-4A shall not exceed 14 days.~~

~~4) The time period to accommodate replacement of the chilled water piping in the MER-3 associated with chiller 1-VS-E-4C shall not exceed 14 days.~~

4.9 RADIOACTIVE GAS STORAGE MONITORING SYSTEM

Applicability

Applies to the periodic monitoring of radioactive gas storage.

Objective

To ascertain that waste gas is stored in accordance with Specification 3.11.

Specification

A. The concentration of oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.11.A by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitor required to be OPERABLE by Table 3.7-5(a) of Specification 3.7.~~E~~D.

B. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.11.B at the frequency specified in the Surveillance Frequency Control Program when the specific activity of the primary reactor coolant is ≤ 2200 μCi/gm dose equivalent Xe-133. Under the conditions which result in a specific activity > 2200 μCi/gm dose equivalent Xe-133, the waste gas decay tanks shall be sampled once per day.

2. Unit Staff

The unit staff organization shall include the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition for each unit as shown in Table 6.1-1.

b. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the position.

c. All core alterations shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during this operation.

Superintendent

d. The operations manager shall hold (or have previously held) a Senior Reactor Operator License for Surry Power Station or a similar design Pressurized Water Reactor plant. The ~~Supervisor~~ Nuclear Shift Operations shall hold an active Senior Reactor Operator License for Surry Power Station.

e. Procedures will be established to insure that NRC policy statement guidelines regarding working hours established for employees are followed. In addition, procedures will provide for documentation of authorized deviations from those guidelines and that the documentation is available for NRC review.

6.1.3 Unit Staff Qualifications

1. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI 3.1 (12/79 Draft) for comparable positions. Exceptions to this requirement are specified in the QA Program. Incumbents in the positions of Shift Manager, Unit Supervisor (SRO), Control Room Operator (RO), and the individual providing advisory technical support to the unit operations shift crew, shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4).

2. For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed Reactor Operator are those individuals who, in addition to meeting the requirements of TS 6.1.3.1 perform the functions described in 10 CFR 50.54(m).

6.7 Environmental Qualifications

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of Licence Nos. DPR-32 and DPR-37 dated October 24, 1980.

no ~~67~~

B. By ~~on~~ later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient details to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

Attachment 3

PROPOSED SUBSEQUENT RENEWED OPERATING LICENSES, TECHNICAL SPECIFICATIONS, AND BASIS PAGES (TYPED)

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

(3) Actions to minimize release to include consideration of:

a. Water spray scrubbing
b. Dose to onsite responders

R. Deleted by Amendment XXX.

S. Deleted by Amendment XXX.

T. Deleted by Amendment XXX.

U. Deleted by Amendment No. 289

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach.)

W. Subsequent Renewed License Conditions

(1) The information in the Updated Final Safety Analysis Report (UFSAR) supplement submitted pursuant to 10 CFR 54.21(d), as revised during the subsequent license renewal application review process, and Virginia Electric and Power Company commitments as listed in Appendix A of the "Safety Evaluation Report Related to the Subsequent License Renewal of Surry Power Station, Units 1 and 2," dated March 2020, are collectively the "Subsequent License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, Virginia Electric and Power Company may make changes to the programs, activities, and commitments described in the Subsequent License Renewal UFSAR Supplement, provided Virginia Electric and Power Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(2) The Subsequent License Renewal UFSAR Supplement, as defined in subsequent renewed license condition (W)(1) above, describes programs to be implemented and activities to be completed prior to the subsequent period of extended operation, which is the period following the May 25, 2032, expiration of the initial renewed license.

a. Virginia Electric and Power Company shall implement those new programs and enhancements to existing programs no later than 6 months before the subsequent period of extended operation.

b. Virginia Electric and Power Company shall complete those activities by the 6-month date prior to the subsequent period of extended operation or by the end of the last refueling outage before the subsequent period of extended operation, whichever occurs later.

c. Virginia Electric and Power Company shall notify the NRC in writing within 30 days after having accomplished item (2)a above and include the status of those activities that have been or remain to be completed in item (2)b above.

4. This subsequent renewed license is effective as of the date of issuance and shall expire at midnight on May 25, 2052.

FOR THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Signed by Veil, Andrea on 05/04/21
Andrea Veil, Director
Office of Nuclear Reactor Regulation

Enclosure:

Appendix A - Technical Specifications for Surry Power Station, Units 1 and 2

Date of Issuance: May 4, 2021

(3) Actions to minimize release to include consideration of:

a. Water spray scrubbing
b. Dose to onsite responders

R. Deleted by Amendment XXX.

S. Deleted by Amendment XXX.

T. Deleted by Amendment XXX.

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V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment
(PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization
method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

W. <u>Subsequent Renewed License Conditions</u>

(1) The information in the Updated Final Safety Analysis Report (UFSAR) supplement submitted pursuant to 10 CFR 54.21(d), as revised during the subsequent license renewal application review process, and Virginia Electric and Power Company commitments as listed in Appendix A of the "Safety Evaluation Report Related to the Subsequent License Renewal of Surry Power Station,
Units 1 and 2," dated March 2020, are collectively the "Subsequent License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, Virginia Electric and Power Company may make changes to the programs, activities, and commitments described in the Subsequent License Renewal UFSAR Supplement, provided Virginia Electric and Power Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(2) The Subsequent License Renewal UFSAR Supplement, as defined in subsequent renewed license condition (W)(1) above, describes programs to be implemented and activities to be completed prior to the subsequent period of extended operation, which is the period following the January 29, 2033, expiration of the initial renewed license.

a. Virginia Electric and Power Company shall implement those new programs and enhancements to existing programs no later than 6 months before the subsequent period of extended operation.

b. Virginia Electric and Power Company shall complete those activities by the 6-month date prior to the subsequent period of extended operation or by the end of the last refueling outage before the subsequent period of extended operation, whichever occurs later.

c. Virginia Electric and Power Company shall notify the NRC in writing within 30 days after having accomplished item (2)a above and include the status of those activities that have been or remain to be completed in item (2)b above.

4. This subsequent renewed license is effective as of the date of issuance and shall expire at midnight on January 29, 2053.

FOR THE UNITED STATES NUCLEAR
REGULATORY COMMISSION

Signed by Veil, Andrea on 05/04/21
Andrea Veil, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A - Technical Specifications for Surry Power Station, Units 1 and 2
Date of Issuance: May 4, 2021

5. REACTOR CRITICAL

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

6. POWER OPERATION

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

7. REFUELING OPERATION

Any operation involving movement of core components when the vessel head is unbolted or removed.

D. OPERABLE

A system, subsystem, train, component, or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). The system or component shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

E. PROTECTIVE INSTRUMENTATION LOGIC

1. ANALOG CHANNEL

An arrangement of components and modules as required to generate a single protective action digital signal when required by a unit condition. An analog channel loses its identity when single action signals are combined.

C. RCS Operational LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. RCS operational LEAKAGE shall be limited to:

 a. No pressure boundary LEAKAGE,

 b. 1 gpm unidentified LEAKAGE,

 c. 10 gpm identified LEAKAGE, and

 d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

2.a. If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.

 b. If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3. If RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within the limit specified in 3.1.C.1.d, the unit shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

10. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

    This restriction does not apply to the movement of the transfer canal door.

11. Two Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS) trains shall be OPERABLE.

    a. With one required train inoperable for reasons other than an inoperable MCR/ESGR envelope boundary, restore the inoperable train to OPERABLE status within 7 days. If the inoperable train is not returned to OPERABLE status within 7 days, comply with Specification 3.10.C.

    b. If two required trains are inoperable or one or more required trains are inoperable due to an inoperable MCR/ESGR envelope boundary, comply with Specification 3.10.C.

12. Manual actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation shall be OPERABLE as specified in TS 3.7.F.

13. Three chillers shall be OPERABLE in accordance with the power supply requirements of Specification 3.23.A. With one of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.A.1, return the inoperable chiller to OPERABLE status within 7 days or comply with Specification 3.10.C. With two of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.A.1, comply with Specification 3.10.C.

14. Eight air handling units (AHUs) shall be OPERABLE in accordance with the operability requirements of Specification 3.23.A. With two AHUs inoperable on the shutdown unit, ensure that one AHU is OPERABLE in each unit's main control room and emergency switchgear room, and restore an inoperable AHU to OPERABLE status within 7 days, or comply with Specification 3.10.C. With more than two AHUs inoperable, comply with Specification 3.10.C.

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

   This restriction does not apply to the movement of the transfer canal door.

3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two MCR/ESGR EVS trains shall be OPERABLE.

   a. With one required train inoperable for reasons other than an inoperable MCR/ESGR envelope boundary, restore the inoperable train to OPERABLE status within 7 days. If the inoperable train is not returned to OPERABLE status within 7 days, comply with Specification 3.10.C.

   b. If two required trains are inoperable or one or more required trains are inoperable due to an inoperable MCR/ESGR envelope boundary, comply with Specification 3.10.C.

5. Manual actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation shall be OPERABLE as specified in TS 3.7.F.

6. Three chillers shall be OPERABLE in accordance with the power supply requirements of Specification 3.23.A. With one of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.A.1, return the inoperable chiller to OPERABLE status within 7 days or comply with Specification 3.10.C. With two of the required OPERABLE chillers inoperable or not powered as required by Specification 3.23.A.1, comply with Specification 3.10.C.

7. Eight air handling units (AHUs) shall be OPERABLE in accordance with the operability requirements of Specification 3.23.A. With two AHUs inoperable on either unit, ensure that one AHU is OPERABLE in each unit's main control room and emergency switchgear room, and restore an inoperable AHU to OPERABLE status within 7 days, or comply with Specification 3.10.C. With more than two AHUs inoperable on a unit, comply with Specification 3.10.C.

C. If any one of the specified limiting conditions for refueling is not met, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease and irradiated fuel shall be placed in a safe position, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

D. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

E. The requirements of 3.0.1 are not applicable.

Basis

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Potential escape paths for fission product radioactivity within containment are required to be closed or capable of closure to prevent the release to the environment. However, since there is no potential for significant containment pressurization during refueling, the Appendix J leakage criteria and tests are not applicable.

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch must be capable of being closed.

2. If a primary source is not available, the unit may be operated for seven (7) days provided the dependable alternate source can be OPERABLE within 8 hours. If specification A-4 is not satisfied within seven (7) days, the unit shall be brought to COLD SHUTDOWN.

3. One battery may be inoperable for 24 hours provided the other battery and battery chargers remain OPERABLE with one battery charger carrying the DC load of the failed battery's supply system. If the battery is not returned to OPERABLE status within the 24 hour period, the reactor shall be placed in HOT SHUTDOWN. If the battery is not restored to OPERABLE status within an additional 48 hours, the reactor shall be placed in COLD SHUTDOWN.

4. One buried fuel oil storage tank may be inoperable for 7 days for tank inspection and related repair, provided the following actions are taken:

   a. prior to removing the tank from service, verify that 50,000 gallons of replacement fuel oil is available offsite and transportation is available to deliver that volume of fuel oil within 48 hours, and

   b. prior to removing the tank from service and at least once every 12 hours, verify that the remaining buried fuel oil storage tank contains ≥ 17,500 gallons, and

   c. prior to removing the tank from service and at least once every 12 hours, verify that the above ground fuel oil storage tank contains ≥ 50,000 gallons.

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.

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

2. Air Handling Units (AHUs)

   a. Unit 1 air handling units, 1-VS-AC-1, 1-VS-AC-2, 1-VS-AC-6, and 1-VS-AC-7, must be OPERABLE whenever Unit 1 is above COLD SHUTDOWN.

      1. If either any single Unit 1 AHU or two Unit 1 AHUs on the same chilled water loop (1-VS-AC-1 and 1-VS-AC-7 or 1-VS-AC-2 and 1-VS-AC-6) become inoperable, restore operability of the one inoperable AHU or two inoperable AHUs within seven (7) days or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      2. If two Unit 1 AHUs on different chilled water loops and in different air conditioning zones (1-VS-AC-1 and 1-VS-AC-6 or 1-VS-AC-2 and 1-VS-AC-7) become inoperable, restore operability of the two inoperable AHUs within seven (7) days or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      3. If two Unit 1 AHUs in the same air conditioning zone (1-VS-AC-1 and 1-VS-AC-2 or 1-VS-AC-6 and 1-VS-AC-7) become inoperable, restore operability of at least one Unit 1 AHU in each air conditioning zone (1-VS-AC-1 or 1-VS-AC-2 and 1-VS-AC-6 or 1-VS-AC-7) within one (1) hour or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

      4. If more than two Unit 1 AHUs become inoperable, restore operability of at least one Unit 1 AHU in each air conditioning zone (1-VS-AC-1 or 1-VS-AC-2 and 1-VS-AC-6 or 1-VS-AC-7) within one (1) hour or bring Unit 1 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

   b. Unit 2 air handling units, 2-VS-AC-8, 2-VS-AC-9, 2-VS-AC-6, and 2-VS-AC-7 must be OPERABLE whenever Unit 2 is above COLD SHUTDOWN.

      1. If either any single Unit 2 AHU or two Unit 2 AHUs on the same chilled water loop (2-VS-AC-7 and 2-VS-AC-9 or 2-VS-AC-6 and 2-VS-AC-8) become inoperable, restore operability of the one inoperable AHU or two inoperable AHUs within seven (7) days or bring Unit 2 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

4.9 RADIOACTIVE GAS STORAGE MONITORING SYSTEM

Applicability

Applies to the periodic monitoring of radioactive gas storage.

Objective

To ascertain that waste gas is stored in accordance with Specification 3.11.

Specification

A. The concentration of oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.11.A by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitor required to be OPERABLE by Table 3.7-5(a) of Specification 3.7.D.

B. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.11.B at the frequency specified in the Surveillance Frequency Control Program when the specific activity of the primary reactor coolant is ≤ 2200 μCi/gm dose equivalent Xe-133. Under the conditions which result in a specific activity > 2200 μCi/gm dose equivalent Xe-133, the waste gas decay tanks shall be sampled once per day.

2. Unit Staff

The unit staff organization shall include the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition for each unit as shown in Table 6.1-1.

b. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the position.

c. All core alterations shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during this operation.

d. The operations manager shall hold (or have previously held) a Senior Reactor Operator License for Surry Power Station or a similar design Pressurized Water Reactor plant. The Superintendent Nuclear Shift Operations shall hold an active Senior Reactor Operator License for Surry Power Station.

e. Procedures will be established to insure that NRC policy statement guidelines regarding working hours established for employees are followed. In addition, procedures will provide for documentation of authorized deviations from those guidelines and that the documentation is available for NRC review.

6.1.3 Unit Staff Qualifications

1. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI 3.1 (12/79 Draft) for comparable positions. Exceptions to this requirement are specified in the QA Program. Incumbents in the positions of Shift Manager, Unit Supervisor (SRO), Control Room Operator (RO), and the individual providing advisory technical support to the unit operations shift crew, shall meet or exceed the requirements of 10 CFR 55.59(c) and 55.31(a)(4).

2. For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed Reactor Operator are those individuals who, in addition to meeting the requirements of TS 6.1.3.1 perform the functions described in 10 CFR 50.54(m).

6.7 Environmental Qualifications

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of Licence Nos. DPR-32 and DPR-37 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient details to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.