UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 10, 2023

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 309 AND 309, RE: LICENSE AMENDMENT REQUEST REGARDING A 10-DAY ALLOWED OUTAGE TIME FOR OPPOSITE UNIT AUXILIARY FEEDWATER CROSS-CONNECT CAPABILITY (EPID: L-2022-LLA-0089)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 309 to Subsequent Renewed Facility Operating License No. DPR-32 and Amendment No. 309 to Subsequent Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry, SPS), Units 1 and 2, respectively. The amendments revise the technical specifications (TSs) in response to your application dated June 20, 2022, as supplemented by letters dated August 8 and October 27, 2022.

The amendments revise TSs Section 3.6.I.2 by permanently extending the allowed outage time (i.e., completion time) from 72 hours to 10 days for the opposite unit Auxiliary Feedwater (AFW) pump cross-connect capability specific to repair/replacement activities and when maintenance that would result in the inoperability of all three of the opposite unit's AFW pumps is being performed.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

John Klos, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 309 to DPR-32
2. Amendment No. 309 to DPR-37
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 309
Subsequent Renewed License
No. DPR-32

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 20, 2022, as supplemented by letters dated August 8 and October 27, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of the Subsequent Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2023.03.10 14:50:40 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Subsequent Renewed Facility
Operating License No. DPR-32
and Technical Specifications

Date of Issuance: March 10, 2023

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 309
Subsequent Renewed License
No. DPR-37

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 20, 2022, as supplemented by letters dated August 8 and October 27, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of the Subsequent Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2023.03.10 14:51:31 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Subsequent Renewed Facility
Operating License No. DPR-37
and Technical Specifications

Date of Issuance: March 10, 2023

ATTACHMENT

SURRY POWER STATION, UNIT NOS. 1 AND 2

AMENDMENT NO. 309 TO

SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

AMENDMENT NO. 309 TO

SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Subsequent Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. DPR-32, Page 3	License No. DPR-32, Page 3
License No. DPR-37, Page 3	License No. DPR-37, Page 3
TSs	TSs
3.6-4a	3.6-4a
3.6-4b	3.6-4b
3.6-5a	3.6-5a
3.6-5b	3.6-5b
-----	3.6-5c

3. This subsequent renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

    A. Maximum Power Level

    The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 309 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

    C. Reports

    The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

    D. Records

    The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

    E. Deleted by Amendment 65

    F. Deleted by Amendment 71

    G. Deleted by Amendment 227

    H. Deleted by Amendment 227

3. This subsequent renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 309 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 54

   F. Deleted by Amendment 59 and Amendment 65

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

I. The requirements of Specification 3.6.C.4 above concerning the opposite unit's auxiliary feedwater pumps; the associated redundant flowpaths, including piping, headers, valves, and control board indication; the cross-connect piping from the opposite unit; and the protected condensate storage tank may be modified to allow the following components to be inoperable, provided immediate attention is directed to making repairs. Automatic initiation instrumentation associated with the opposite unit's auxiliary feedwater pumps need not be OPERABLE.

1. One of the opposite unit's flowpaths or two of the opposite unit's auxiliary feedwater pumps may be inoperable for a period not to exceed 14 days.

2. Both of the opposite unit's flowpaths; the opposite unit's protected condensate storage tank; the cross-connect piping from the opposite unit; or three of the opposite unit's auxiliary feedwater pumps may be inoperable for a period not to exceed 72 hours. For the specific purpose of performing maintenance related to the operability of all three of the opposite unit's auxiliary feedwater pumps, these components may be inoperable for a period not to exceed 10 days.*

3. A train of the opposite unit's emergency power system as required by Section 3.6.C.4.c above may be inoperable for a period not to exceed 14 days; if this train's inoperability is related to a diesel fuel oil path, one diesel fuel oil path may be "inoperable" for 24 hours provided the other flowpath is proven OPERABLE; if after 24 hours, the inoperable flowpath cannot be restored to service, the diesel shall be considered "inoperable." During this 14 day period, the following limitations apply:

   a. If the offsite power source becomes unable to energize the opposite unit's OPERABLE train, operation may continue provided its associated emergency diesel generator is energizing the OPERABLE train.

---

* The compensatory measures identified in the letter from Virginia Electric and Power Company to the US NRC dated June 20, 2022 (ADAMS Accession No. ML22172A134) and listed in the TS 3.6 Basis are required to be in place whenever the 10-day Allowed Outage Time is entered.

b. If the opposite unit's OPERABLE train's emergency diesel generator becomes unavailable, operation may continue for 72 hours provided the offsite power source is energizing the opposite unit's OPERABLE train.

c. Return of the originally inoperable train to OPERABLE status allows the second inoperable train to revert to the 14 day limitation.

If the above requirements are not met, be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours.

J. The requirements of Specification 3.6.C.2 above may be modified to allow utilization of protected condensate storage tank water with the auxiliary feedwater pumps provided the water level is maintained above 60,000 gallons, sufficient replenishment water is available in the 300,000 gallon condensate storage tank, and replenishment of the protected condensate storage tank is commenced within two hours after the cessation of protected condensate storage tank water consumption.

restore operability of one inoperable pump or of the inoperable component or instrumentation in one flowpath. With such a loss of auxiliary feedwater capability, the unit is in a seriously degraded condition. In this condition, the unit should not be perturbed by any action, including a power change, which could result in a plant transient or trip. The seriousness of this condition requires that action be taken immediately to restore operability, where immediately means the required action should be pursued without delay and in a controlled manner. Under these circumstances, Specification 3.0.1 and all other required actions directing mode changes are suspended until one inoperable pump or the inoperable component or instrumentation in one flowpath is restored to operable status, because taking those actions could place the unit in a less safe condition.

Due to the occasional need to perform maintenance on common AFW components that would render all three AFW pumps on the opposite unit inoperable, e.g., the AFW pumps' full flow recirculation piping or the protected condensate storage tank, a 10-day allowed outage time is provided for opposite unit AFW cross-connect capability. The 10-day allowed outage time is supported by a risk analysis that demonstrates the associated risk is acceptably small for both CDF and LERF with considerable margin remaining. When entering the 10-day allowed outage time for the specific purpose of performing maintenance related to the operability of all three of the opposite unit's auxiliary feedwater pumps, the following compensatory measures are required to be in place:

- Additional AFW system maintenance, including associated water sources, or changes in plant configuration that would result in a risk significant configuration will be precluded;
- Weather conditions will be monitored and AFW maintenance affecting operability of the opposite unit cross-connect will not be scheduled if severe weather conditions are anticipated;
- The steam-driven AFW pump will be controlled as “Protected Equipment”;
- The Technical Requirements Manual compensatory actions to address 10 CFR 50.65 (a)(4) fire risk related to AFW cross-connect unavailability, which include periodic walkdowns in relevant fire areas, will be taken; and
- The BDB/FLEX AFW pump will be pre-staged to provide AFW defense-in-depth comparable to the AFW cross-connect.

In the event of complete loss of electrical power to the station, residual heat removal would continue to be assured by the availability of either the turbine driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000-gallon protected condensate storage tank.

In the event of a fire or high energy line break which would render the auxiliary feedwater pumps inoperable on the affected unit, residual heat removal would continue to be assured by the availability of either the turbine driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps from the opposite unit. A minimum of two auxiliary feedwater pumps are required to be operable* on the opposite unit to ensure compliance with the design basis accident analysis assumptions, in that auxiliary feedwater can be delivered via the cross-connect, even if a single active failure results in the loss of one of the two pumps. In addition, the requirement for operability of the opposite unit's emergency power system is to ensure that auxiliary feedwater from the opposite unit can be supplied via the cross-connect in the event of a common-mode failure of all auxiliary feedwater pumps in the affected unit due to a high energy line break in the main steam valve house. Without this requirement, a single failure (such as loss of the shared backup diesel generator) could result in loss of power to the opposite unit's emergency buses in the event of a loss of offsite power, thereby rendering the cross-connect inoperable. The longer allowed outage time for the opposite unit's emergency power system is based on the low probability of a high energy line break in the main steam valve house coincident with a loss of offsite power.

The specified minimum water volume in the 110,000-gallon protected condensate storage tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all offsite electrical power. If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000-gallon condensate tank can be gravity-fed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

---

* excluding automatic initiation instrumentation

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,842,454 pounds per hour at their individual relieving pressure; the total combined capacity of all fifteen main steam code safety valves is 11,527,362 pounds per hour. The maximum steam flow at full power is approximately 11,444,000 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady state power than can be obtained during three reactor coolant loop operation.

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine-131 activity is based on limiting the dose at the site boundary following a postulated steam line break accident to the Regulatory Guide 1.183 limits. The accident analysis assumes the release of the entire contents of the faulted steam generator to the atmosphere.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 309 TO SUBSEQUENT RENEWED FACILITY

OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 309 TO SUBSEQUENT RENEWED FACILITY

OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated June 20, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22172A134), as supplemented by letters dated August 8 (ML22220A216) and October 27, 2022 (ML22300A188), Virginia Electric and Power Company (VEPCO, the licensee) submitted a license amendment request (LAR) for changes to the Surry Power Station (Surry, SPS), Units 1 and 2, technical specifications (TSs).

The requested changes would revise TS Section 3.6.I.2 by permanently extending the allowed outage time (AOT) (i.e., completion time (CT)) from 72 hours to 10-days for opposite unit Auxiliary Feedwater System (AFW) pumps cross-connect capability specific to repair/replacement activities and when maintenance that would result in the inoperability of all three of the opposite unit's AFW pumps is being performed.

The supplements dated August 8 and October 27, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 6, 2022 (87 FR 54555).

2.0 REGULATORY EVALUATION

2.1 Auxiliary Feedwater Description

In its letter dated June 20, 2022, the licensee provided a description of the AFW system in Section 2.1, "System Design and Operation," that states, in part, that:

> The SPS AFW system (see Figure 1) provides a source of feedwater to the secondary side of the steam generators (SGs) at times when the Main Feedwater (MFW) system is not available, thereby maintaining the heat sink capabilities of the SGs. The system is relied upon to prevent core damage and Reactor Coolant System (RCS) over pressurization in the event of transients, such as a loss-of-normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.
>
> The AFW system for each unit consists of two motor-driven AFW pumps, each rated for 350 gallons per minute (gpm) at 2730 feet of head, one steam-driven AFW pump rated for 700 gpm at 2730 feet of head, a 110,000 gallon Emergency Condensate Storage Tank (ECST), and associated piping, headers, valves, controls, and instrumentation.

2.2 Proposed TS Change

The licensee proposes to modify TS 3.6.I.2 to extend the AOT for an inoperable opposite unit AFW pump system from 72 hours to 10 days. This proposed change would allow for an extended AOT for opposite unit cross-connect capability due to maintenance activities related to the operability of all three of the opposite unit's AFW pumps to facilitate necessary unit specific common AFW pipe repairs, ECST maintenance, etc., without requiring the shutdown of the operating unit.

The licensee stated that the proposed change is needed to replace the Unit 2 AFW full flow recirculation piping. The licensee determined that this task could not be completed within the TS 3.6.I.2 AOT for the cross-connect and would require taking Unit 1 offline to complete the Unit 2 repair.

In its letter dated October 27, 2022, the licensee proposed TS 3.6.I.2 to be revised as follows (change indicated in **bold text):**

> Both of the opposite unit's flowpaths; the opposite unit's protected condensate storage tank; the cross-connect piping from the opposite unit; or three of the opposite unit's auxiliary feedwater pumps may be inoperable for a period not to exceed 72 hours. **For the specific purpose of performing maintenance related to the operability of all three of the opposite unit's auxiliary feedwater pumps, these components may be inoperable for a period not to exceed 10 days.***

In its letter dated October 27, 2022, the licensee also proposed the accompanying footnote:

> *** The compensatory measures identified in the letter from Virginia Electric and Power Company to the US NRC dated June 20, 2022 (ADAMS Accession No. ML22172A134) and listed in the TS 3.6 Basis are required to be in place whenever the 10-day Allowed Outage Time is entered.**

In its letter dated June 20, 2022, the licensee proposed to add the following to the TS 3.6 Basis:

> Due to the occasional need to perform maintenance on common AFW components that would render all three AFW pumps on the opposite unit inoperable, e.g., the AFW pumps' full flow recirculation piping or the protected condensate storage tank, a 10-day allowed outage time is provided for opposite unit AFW cross-connect capability. The 10-day allowed outage time is supported by a risk assessment that demonstrates the associated risk is acceptably small for both CDF[core damage frequency and LERF [large early release frequency] with considerable margin remaining. When entering the 10-day allowed outage time for the specific purpose of performing maintenance related to the operability of all three of the opposite unit's auxiliary feedwater pumps, the following compensatory actions are required to be in place:
>
> - Additional AFW system maintenance, including associated water sources, or changes in plant configuration that would result in a risk significant configuration will be precluded;
> - Weather conditions will be monitored and AFW maintenance affecting operability of the opposite unit cross-connect will not be scheduled if severe weather conditions are anticipated;
> - The steam-driven AFW pump will be controlled as "Protected Equipment";
> - The Technical Requirements Manual compensatory actions to address 10 CFR 50.65 (a)(4) fire risk related to AFW cross-connect unavailability, which include periodic walkdowns in relevant fire areas, will be taken; and
> - The Beyond Design Basis BDB/FLEX AFW pump will be pre-staged to provide AFW defense-in-depth comparable to the AFW cross-connect.

The NRC staff considered the proposed TS Basis changes and compensatory measures as part of its review, but NRC does not approve TS Basis changes as part of its TS approval.

2.3 Regulatory Requirements and Guidance

The regulatory requirements and guidance on which the NRC staff bases its acceptance of risk-informed AOT extension requests are:

The regulations in Title 10 of the *Code of Federal Regulations (*10 CFR 50.90), "Application for amendment of license, construction permit, or early site permit," state, in part, that whenever a holder of a license desires to amend the license, application for amendment must be filed with the Commission fully describing the changes desired.

The regulations in 10 CFR 50.92(a), “Issuance of amendment,” state, in part, that

> …the Commission will be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate…

The regulations in 10 CFR 50.36(2)(b), “Technical specifications,” state, in part, that

> …The technical specifications will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34...

The regulations in 10 CFR 50.36(c)(2), “Technical specifications, - *Limiting conditions for operation*,” state, in part, that TSs will include

> (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

In determining whether the proposed TS remedial actions should be granted, the Commission will apply the “reasonable assurance” standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation in 10 CFR 50.40(a) states, in part, that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether:

> ...the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other TSs, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in part 20 of this chapter, and that the health and safety of the public will not be endangered.

The regulation at 10 CFR 50.57(a)(3) states, in part, that the Commission may issue an operating license amendment when:

> (3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the regulations in this chapter.

Regulatory Guide (RG) 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018 (ML17317A256), describes an approach that is acceptable to the NRC for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights.

RG 1.177, Revision 2, “Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” January 2021 (ML20164A034), describes an approach acceptable to the NRC staff for developing risk-informed applications for changes to the completion times (CTs) and surveillance frequencies (SFs) of plant TS. Section C.2.4, “Acceptance Guidelines for Technical

Specification Changes," of RG 1.177 provides the following three-tiered TS acceptance guidelines for evaluating the risk associated with CT changes:

- The licensee has demonstrated that the TS CT change has only a small quantitative impact on plant risk. An ICCDP [incremental conditional core damage probability] of less than 1×10^{-6} and an ICLERP [incremental conditional large early release probability] of less than 1×10^{-7} are considered small for a single TS condition entry (Tier 1).

- The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2).

- The licensee has implemented a risk-informed plant configuration control program, including procedures to use, maintain, and control such a program (Tier 3).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ML090410014), describes one acceptable approach for determining whether the technical adequacy of the probabilistic risk assessment (PRA), in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light water reactors.

NUREG-0800, Revision 1, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [Light-Water Reactor] Edition," SRP Chapter 16.1, Revision 1, "Risk-Informed Decision Making: Technical Specifications," March 2007 (ML070380228) provides review guidance to the NRC staff in evaluating the five key principles provided in RG 1.174 and 1.177 that all risk-informed applications are expected to meet.

NUREG-0800, SRP, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load," September 2012 (ML12193A107) provides guidance to the NRC staff on evaluating the technical adequacy, scope and level of detail of the baseline PRA used by the licensee to support license amendments. NUREG-0800, SRP, Section 19.2, Initial Issuance, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ML071700658) provides general guidance to the NRC staff for evaluating the technical basis for proposed risk-informed changes.

3.0 TECHNICAL EVALUATION

3.1 Risk-Informed Evaluation

All risk-informed applications for changes to plant TSs are expected to meet the following five key principles, as provided in RG 1.174, Section C, and RG 1.177, Section C.2:

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12).

Principle 2: The proposed licensing basis change is consistent with the defense-in-depth [DID] philosophy.

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When proposed licensing basis changes results in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants. [“Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” 51 FR 30028 (August 21, 1986)]

Principle 5: The impact of the proposed licensing basis change should be monitored using performance measurement strategies.

3.1.1 Key Principle 1 – NRC Staff Evaluation of Compliance with Current Regulations

The regulation pertinent to the licensee's proposed TS amendment request is 10 CFR 50.36(c)(2)(ii), Criterion 3, that states that “A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” The licensee proposed a permanent change to TS 3.6.I.2 to increase the AOT from 72 hours to 10 days for opposite unit cross-connect capability due to the occasional need to perform maintenance on the AFW pump system that would affect the operability of all three of the opposite unit's AFW pumps. The unavailability of the AFW pumps cross-connect will be limited to the 10 days duration for the specific purpose of repair/replacement activities and for future maintenance. No exemption was requested.

The requested change does not propose any deviation or exemption to the regulation itself, but requests a permanent change concerning how the TS is implemented within the regulation. Based on the above, the NRC staff concludes that the proposed change would continue to meet 10 CFR 50.36 and is, therefore, acceptable.

3.1.2 Key Principle 2 – NRC Staff Evaluation of Defense-in-Depth Attributes

In its letter dated August 8, 2022, the licensee stated:

> The proposed change is consistent with the defense-in-depth philosophy. The SPS AFW system is a diverse system that has two independent trains of motor-driven pumps and a third train with a steam-driven pump. In most scenarios, the likelihood of losing all three trains of one unit's AFW system is very low. The SPS AFW cross-connect provides defense-in-depth during a loss of a unit's AFW and provides decay heat removal during a fire or high energy line break in the main steam valve house where the AFW pumps are located because those scenarios can impact all three AFW pumps for one unit. Under the proposed change, a Beyond Design Basis (BDB)/FLEX pump capable of providing AFW flow to the affected unit will be pre-staged for use if needed whenever the 10-day AOT is entered. Since the diesel-driven BDB/FLEX pump can provide AFW flow for decay heat removal in all of the scenarios where the AFW pumps cross-connect would normally be used, the normal level of defense-in-depth is preserved for all relevant accident scenarios.

Defense-in-Depth (DID) is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. DID includes the use of access controls, physical barriers, redundant, and diverse key safety functions, and emergency response measures. Defense-in-depth is often characterized by varying layers of defense, each of which may represent conceptual attributes of nuclear power plant design and operation or tangible objects such as the physical barriers between fission products and the environment.

As discussed throughout RG 1.174, consistency with the DID philosophy is maintained by the following measures:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential CCFs [common-cause failures].
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

The staff has reviewed the information provided in the LAR and the final safety analysis report against the defense-in-depth attributes discussed in RG 1.174, and finds that:

<u>Preserve a reasonable balance among the layers of defense</u>

The design basis of the AFW is to supply water to the steam generator to remove decay and other residual heat. The licensee is proposing no change to the design of the plant or any operating parameters, no new operating configurations, and no new changes to the design basis in the proposed tech spec change. Therefore, the NRC staff finds that a reasonable balance among the layers of defense is preserved.

<u>Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures</u>

The proposed change will not involve any new programmatic activities or credit any new operator actions for assuring the AFW can perform its safety function during normal operation. However, voluntary compensatory measures that involve making flex equipment available is included to provide defense-in-depth for repair/replacement or maintenance activities during the temporary plant configuration and provide cooling to the opposite unit to mitigate the potential for a common-cause failure in the main steam valve area. The use of this compensatory measure provides added capacity above the original design and licensing basis and does not reduce the capability of AFW design features for any of the units. The licensee has also demonstrated that the impact on plant risk for this proposed change is acceptable because an ICCDP of less than 1×10-6 and an ICLERP of less than 1x10-7 with effective compensatory measures will be implemented to reduce the sources of increased risk. Therefore, the NRC staff finds that the proposed TS change preserves adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty

The licensee is proposing no change to the design of the plant or any operating parameters, and no new changes to the design basis in the proposed TS change. The addition of a 10-day AOT for opposite unit AFW cross-connect capability is needed due to the need to perform repair/replacement and the occasional need to perform maintenance on the AFW system that would affect the operability of all three of the opposite unit's AFW pumps. Completion of repair/replacement and maintenance in the configuration for the 10-day LCO would be a rare circumstance and configuration needed to maintain redundancy, independence, and diversity. Therefore, the NRC staff finds that system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system is preserved.

Preserve adequate defense against potential common-cause failures

The cross-connect ensures that auxiliary feedwater from the opposite unit can be supplied via the cross-connect in the event of a common-mode failure of all auxiliary feedwater pumps in the affected unit due to a high energy line break in the main steam valve house. When the cross-connect AFW system is temporarily unavailable, compensatory measure will be put into place to ensure that an alternative for providing feedwater to the opposite unit's SGs is capable of being used to remove decay heat. The compensatory measure will make use of pre-staged BDB/FLEX pumps to provide alternate cooling of the opposite unit SG if needed prior to entry into the 10-day AOT, as specified in the proposed Note included in TS 3.6.I.2. Therefore, the NRC staff finds that adequate defense against potential CCFs are preserved.

Maintain multiple fission product barriers

The relationship of AFW system to individual barriers will not change as a result of the new AOT. The licensee is providing BDB/FLEX measures to enhance defense-in-depth for secondary cooling to the steam generators that provide heat removal for the primary system and therefore, serves to maintain integrity of the reactor coolant system as a fission product barrier. Therefore, the NRC staff finds that multiple fission product barriers protection is unaffected and maintained.

Preserve sufficient defense against human errors

Operators' response during normal, abnormal, and emergency operating conditions will continue to be in accordance with station approved procedures. Dedicated and trained operators are stationed in key areas to assure that specified cooling functions are performed in the time required. Therefore, the NRC staff finds that sufficient defense against human error will continue to be preserved.

Continue to meet the intent of the plant's design criteria

The requested change does not result in any design or physical changes to the AFW system. Therefore, the NRC staff finds that plant will continue to meet the intent of the plant's design criteria.

3.1.3 Key Principle 3 – NRC Staff Evaluation of Safety Margins

Section 2.2.2 of RG 1.177, Revision 2, states, in part, that sufficient safety margins are maintained when:

> Codes and standards or alternatives approved for use by the NRC are met.
>
> Safety analysis acceptance criteria in the Updated Final Safety Analysis Report are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

The licensee is not proposing to change any quality standard, material, or operating specifications. Acceptance criteria for operability of equipment are not changed and use of the extended CT only affects the AFW pump cross-connect capability specific to repair/replacement activities and when maintenance that would result in the inoperability of all three of the opposite unit's AFW pumps occurs.

The current license, which includes TS as Appendix A to the licenses, allows for AFW pump systems to be out of service for 72 hours and was issued after the NRC determined there was reasonable assurance of public health and safety, and compliance with NRC regulations. The licensee is not proposing any changes to its design and will continue to meet applicable codes and standards. However, to maintain sufficient safety margins during the additional seven days of unavailability of the AFW pumps, the licensee proposes using compensatory measures described in the LAR. The licensee stated in its LAR that an extended period without the AFW pump cross-connect available requires an alternate method to provide decay heat removal for the scenarios that depend upon the cross-connect. The staging of a BDB/FLEX (diverse and flexible coping strategy) pump that can provide AFW flow to the unit that would normally use the AFW pump cross-connect will ensure safe shutdown capability is maintained during the extended AOT. The availability and pre-staging of the BDB/FLEX pump whenever the 10-day AOT is entered, in addition to the other required compensatory actions discussed in the LAR, will maintain appropriate safety margins at all times. The proposed compensatory measures also reduce the potential for maintenance errors to challenge the proper functioning of equipment to meet their intended safety functions. Therefore, NRC staff finds that the specified safety functions would remain with sufficient safety margins during the 10-day AOT.

3.1.4 Key Principle 4 – NRC Staff Evaluation of Increases in Risk

Principle 4 states that when proposed licensing basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operation of nuclear power plants. The licensee's proposed TS change uses the three-tiered approach described in RG 1.177 to address the calculated change in risk as measured by the change in core damage frequency (ΔCDF), change in large early release frequency (ΔLERF), ICCDP, ICLERP, and the use of compensatory measures to reduce risk.

The NRC staff evaluated the licensee's processes and methodologies for determining that the change in risk from implementing AFW pump system extended AOT will be small and consistent with RG 1.174 and RG 1.177 guidance. The NRC staff evaluated the licensee's proposed changes and methods for determining the risk for a proposed AOT against the three-tiered approach in RG 1.177. The results of the NRC staff's review are discussed below.

The license indicated that the Surry PRA was utilized to evaluate the impact on CDF and LERF to support the application. The AFW pumps are safety-related and shared across Units 1 and 2, and that the cross-connect is modeled explicitly in the average maintenance model for Surry.

Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed change on plant operational risk. The Tier 1 review involves two aspects: (1) evaluation of the technical adequacy of PRA models and their application to the proposed change, and (2) evaluation of the PRA results and insights based on the licensee's proposed change.

PRA Technical Adequacy

Section 2.3 of RG 1.174, Revision 3, states, in part that:

> [t]he PRA analysis used to support an application is measured in terms of its appropriateness with respect to scope, level of detail, conformance with the technical elements, and plant representation. These aspects of the PRA are to be commensurate with its intended use and the role the PRA results play in the integrated decision process.

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

RG 1.200, Revision 2, describes one approach acceptable to the NRC for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for LWRs. RG 1.200, Revision 2, endorses, with exceptions and clarifications, the use of: (1) the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications;” (2) Nuclear Energy Institute (NEI) 00-02, Revision 1, “Probabilistic Risk Assessment (PRA) Peer Review Process Guidance” (ML061510619); and (3) NEI 05-04, Revision 2, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard” (ML083430462). The ASME/ANS PRA standard provides technical supporting requirements in terms of three Capability Categories (CCs). The intent of the delineation of the Capability Categories within the supporting requirements is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from CC-I to CC-III. In general, the NRC staff anticipates that current good practice (i.e., CC-II of the ASME/ANS standard) is adequate for the majority of applications.

In Section 3.1.1.3 of its letter dated June 20, 2022, the licensee stated that a discussion of Surry PRA quality was provided in the license amendment request to adopt 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors,” dated December 6, 2019 (ML19343A019). The NRC staff evaluated the scope of the PRA including: (1) peer review history and results (includes open Findings and Observations (F&Os)); (2) the F&O closure process; (3) credit for FLEX in the PRA; and (4) assessment of assumptions and approximations, and documented the NRC staff’s conclusions regarding PRA quality for the purpose of risk-informed categorization and treatment, pursuant to 10 CFR 50.69, in its safety evaluation report dated December 8, 2020 (ML20293A160).

Internal Events PRA

RG 1.200, Revision 2, Regulatory Position 4.2 states that previously submitted documentation may be referenced if it is acceptable for the subject submittal and that licensees are expected to assure that the PRA model represents the as-designed, as-built, as-operated plant. Identify permanent plant changes (such as design or operational practices) that have an impact on those things modeled in the PRA but have not been incorporated in the base PRA model. If a plant change has not been incorporated, provide a justification of why the change does not impact the PRA results used to support the application. This justification should be in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same.)

In its letter dated October 27, 2022, the licensee responded to request for additional information No. 1 (RAI-1) regarding upgrades made to the PRA models since the approval of the 10 CFR 50.69 LAR. The licensee stated:

> Only one upgrade was made to the SPS PRA model since the approval of the 10 CFR 50.69 LAR. This upgrade applied thermal hydraulic analysis from the GOTHIC containment analysis code to demonstrate the SPS containment would not be at risk of an overpressure failure in certain loss of coolant accident (LOCA) scenarios where the Modular Accident Analysis Program code predicted failure would occur. This model change had a significant impact on the plant risk profile, particularly Large Early Release Frequency (LERF) sequences caused by a small break LOCA.
>
> For the LAR application, the dominant risk scenario is a fire or a steam line break in the main steam valve house, where the operating unit's AFW pumps are located, combined with the unavailability of the AFW cross-connect, which leaves only the Beyond Design Basis (BDB)/FLEX AFW pumps to supply the AFW function. Therefore, LOCA scenarios are not major contributors for this application, and the PRA upgrade doesn't impact the risk insights or the acceptability of this application. The SPS PRA model with this upgrade applied can continue to be used to effectively assess configuration risk associated with the proposed 10-day AOT for opposite unit AFW cross-connect capability.

In Section 3.1.1.3 of its submittal dated June 20, 2022, the licensee stated:

> The NRC staff evaluated the scope of the PRA including: (1) peer review history and results (includes open Findings and Observations (F&Os)); (2) the F&O closure process; (3) credit for mitigating strategies (FLEX) in the PRA; and (4) assessment of assumptions and approximations, and documented their acceptance in the safety evaluation report included in SPS Units 1 and 2 Amendment Nos. 301/301 dated December 8, 2020 (Reference 6.2) [ML20293A160]. The disposition of the six outstanding F&Os as discussed in those documents is applicable to this LAR as well. Included in the model is a correction to a Human Error Probability (HEP) that restarts MFW after a Safety Injection (SI) signal. The HEP needed to be coupled with an SI signal to satisfy the initial conditions for the HEP.

> The six F&Os in the Surry internal events PRA model that are currently open and their impact on this application are described in the table below.

In its table entitled, "Disposition of Currently Open F&Os - Surry Internal Events PRA Model," the licensee indicated that the six open F&Os do not impact the AOT for the opposite unit AFW pump cross-connect capability. The NRC staff reviewed each open F&O and the licensee's disposition to determine whether the F&Os had any significant impact on the application. The NRC staff determined that the open F&Os have no impact on this application.

PRA Results and Insights

In its letter dated June 20, 2022, the licensee provided the table below, to summarize its calculated ΔCDF, ΔLERF, ICCDP, and ICLERP for the proposed AOT extension for the AFW pump system cross-connect.

Table 1: Tier 1 Criteria						
	Unit 1 ΔCDF	Unit 2 ΔCDF	RG 1.177 ΔCDF Criteria	Unit 1 ΔLERF	Unit 2 ΔLERF	RG 1.177 ΔLERF Criteria
10 Days/Year Unavailability	1.15E-08	1.16E-08	1.00E-06	2.35E-09	2.36E-09	1.00E-07
	Unit 1 ICCDP	Unit 2 ICCDP	RG 1.177 ICCDP Criteria	Unit 1 ICLERP	Unit 2 ICLERP	RG 1.177 ICLERP Criteria
Single 10-Day TS Entry	9.64E-09	1.02E-08	1.00E-06	1.99E-09	2.03E-09	1.00E-07

The licensee stated, in part, that, as evidenced in Table 1, extensive margin exists based on the RG 1.177 acceptance criteria.

> The dominant risk scenario is a high energy line break (HELB, either Feedwater (FW) or Main Steam (MS)) in the Unit 1 MSVH [Main Steam Valve House] that could damage the Unit 1 AFW pumps. This scenario would leave Unit 1 without any source of AFW, leaving feed and bleed cooling as the only method of decay heat removal.

The change in risk to extend the AFW pumps system AOT is very small. The licensee provided sufficient evaluation to conclude that the PRA is acceptable to support the risk analysis for the permanent extension of the AOT for the AFW pump system cross-connect. The NRC staff concludes that the proposed risk increase is very small for the requested action and configuration. Based on the above, NRC finds that an additional evaluation of the total risk is not needed because the very small risk values are within the acceptable criteria for RG 1.174, and their acceptability does not depend on a complete reevaluation the of total plant risk.

Internal Fire Hazard Evaluation

The licensee performed a review of its Individual Plant Examination for External Events (IPEEE) and fire contingency action procedures to evaluate the impact for the extension of the AOT on fire risk since a full-scope fire PRA model has not been developed for Surry. In its letter dated June 20, 2022, the licensee stated that the IPEEE identified four areas as significant contributors to the fire CDF. The areas that were identified include the cable vault and tunnel, the emergency switchgear room, the main control room, and the normal switchgear room. Since fires in other areas are not significant contributors to fire risk as characterized by the IPEEE, they are screened from further consideration.

The NRC staff reviewed the Surry IPEEE and reported the results in its March 7, 2000, “Review of Surry Power Station Units 1 & 2 Individual Plant Examination of External Events (IPEEE) Submittal,” (ML003692174). The NRC staff concluded that, “[o]n the basis of the IPEEE review, the NRC staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities.”

In its letter dated June 20, 2022, the licensee stated:

> Input from the SPS IPEEE was used to perform a quantitative risk estimate of the ICCDP for the 10-day AOT from fire hazards. Initiating event frequencies were used from the IPEEE and combined with basic event probabilities from the current SPS internal events model. Adjustments were made to reflect the unavailability of the AFW cross-connect and the ability to use the pre-staged BDB/FLEX AFW pump to mitigate some scenarios. A screening value of 0.1 was used to represent the failure rate of the recovery using the BDB/FLEX pump. The ICCOP for fire hazard is 1.69E-7. The dominant risk frequency is a fire in an ESGR with a failure of the steam-driven AFW pump, failure of the delivery line due to various valve failures, and failure of the BDB/FLEX AFW pump. These sequences lead to core damage. LERF was not quantified for fire risk in the IPEEE, so ICLERP for fire risk was not calculated but is expected to be significantly lower than COF since the limiting scenarios do not directly challenge containment pressure or isolation, and the internal events LERF fractions for these sequences are less than 0.1.
>
> To mitigate fire risk, MRule (a)(4) Fire Risk Equipment AFW cross-connect risk mitigation actions will be followed whenever the 10-day AOT is entered. Based on the review of risk significant fire areas, the expected equipment damage, and the fire strategies used to achieve safe shutdown (SSD), it is concluded the fire risk of having the AFW cross-connect unavailable for 10 days during these scenarios is within the acceptance guidelines in Reg Guide 1.174.

Based on the above, the NRC staff concludes that the qualitative assessment of the significant fire areas provide confidence that the licensee’s use of IPEEE insights demonstrate that the impact of the AFW pumps cross-connect AOT change in risk is acceptable and bounded by the internal events analysis.

<u>Seismic Hazard Evaluation</u>

The seismic PRA quantification in the IPEEE concluded that no severe accident vulnerabilities exist at Surry from a potential seismic event. The NRC staff reviewed the Surry IPEEE and reported the results in its March 7, 2000, “Review of Surry Individual Plant Examination of External Events (IPEEE) Submittal,” (ML003692174). The NRC staff concluded that, “[o]n the basis of the IPEEE review, the NRC staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the IPEEE submittal for the Surry Power Station Units 1 and 2 has met the intent of Supplement 4 to Generic Letter 88-20." The NRC noted that the licensee estimated a seismic CDF of 8E-06/year, using the Electric Power Research Institute hazard curves and a uniform hazard spectrum, and estimated the fire core damage frequency to be 6E-6/year from the fire areas which were not screened out.

The licensee also reviewed the Surry Seismic Probability Risk Assessment Pilot Plant Report for this LAR and concluded that the proposed extended AOT would have minimal impact on the seismic risk. In its letter dated June 20, 2022, the licensee stated:

> Seismic failures are usually correlated meaning that if a seismic event severe enough to fail one AFW pump occurs, then it would more than likely cause all of the AFW pumps to fail. In this case, there is no impact to plant risk from the AFW cross-connect being unavailable for a longer period of time because there would not be any pumps to provide water through the cross-connect. Depending on the event, the AFW function may be provided by the staged or stored BDB/FLEX AFW pumps because seismic failure of these pumps would not be correlated with the installed AFW pumps.

Section 3.1 of RG 1.200, Revision 2, states that missing hazard groups may be evaluated using bounding arguments to cover the risk contributions not addressed by the model. In the LAR dated June 20, 2022, the licensee summarized how it performed the screening analysis referencing acceptable NRC methods. The NRC staff finds that the licensee's use of NRC reviewed methods and models to evaluate the risk impact of the proposed change is sufficient to conclude that the evaluation is adequate to support the proposed change.

<u>Shutdown Risk Evaluation</u>

The licensee assessed shutdown risk using a qualitative evaluation process. In its evaluation, the licensee used its site shutdown risk management procedure, and then considered what impacts the application may have on shutdown DID, in particular, the following shutdown key safety functions: RCS/Spent Fuel Pool Inventory Control, Reactivity Control, Electrical Power, Spent Fuel Pool Cooling, and Containment Integrity. In its letter dated June 20, 2022, the licensee discussed the DID features available in its LAR for the 10-day AOT and stated:

> Since the AFW cross-connect is only the backup to the Natural Circulation requirement for shutdown when the RCS pressure is above 84 psig, it is concluded the proposed change has negligible impact on shutdown core damage frequency (COF) and large early release frequency (LERF). Additionally, a BDB/FLEX pump is pre-staged each outage so it can be used to support an unexpected transient on a shutdown unit. This pump provides an additional layer of defense-in-depth that supports the decay heat removal key safety function when the AFW cross-connect is unavailable.

The NRC staff concludes that the licensee's plan to use site administrative procedures, planned actions, and DID features to ensure that the shutdown risk impact for the extended AOT is appropriately assessed and managed for the proposed permanent TS change request is acceptable.

Other External Hazards Evaluation

In its letter dated June 20, 2022, the licensee stated external hazards as identified by NUREG/CR-2300, (Volume 1, ML063560439, Volume 2, ML063560440), “A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," and NUREG/CR-4839, “Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development,” (ML062260210) have been taken into consideration. The external hazards were evaluated in response to Generic Letter 1988-020, Supplement 4 “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” (ML031150485). The licensee stated:

> Therefore, it can be concluded that non-seismic external events do not pose a significant risk to the safe operation of SPS. With respect to wind hazards, the AFW pumps are located in missile protected buildings, and the other BDB/FLEX AFW pumps will remain in the missile protected dome to backup AFW function if necessary. Based on the region of Virginia where SPS is located, the wind hazard is not high enough to require development of a High Winds PRA. Therefore, based on this evaluation, other External Events have been screened from further consideration in this risk analysis.

Based on the above, the NRC concludes that the licensee’s evaluation provides reasonable assurance that there is no significant external hazards and is, therefore, acceptable.

Tier 2: Avoidance of Risk-Significant Configurations

A licensee is expected to provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service in accordance with the proposed TS change. The avoidance of risk-significant plant configurations limits potentially high-risk configurations that could exist if equipment, in addition to that associated with the proposed TS change, is simultaneously removed from service or other risk-significant operational factors such as concurrent system or equipment testing are involved. Therefore, Tier 2 helps ensure that appropriate restrictions are placed on dominant risk-significant configurations relevant to the proposed TS change.

The licensee performed a cut set review that focused on loss of the AFW pump cross-connect, and no additional risk significant configurations were identified that need to be addressed. In its response to RAI-4 dated October 27, 2022, regarding the licensee’s approach to avoid any risk significant configurations, the licensee described its approach to protecting equipment and ensuring its continued operation through compensatory measures in place including pre-staging of a BDB/FLEX AFW pump. The licensee stated:

> Several compensatory actions identified in this application rely on the availability of key components to backup safety functions if needed. The proposed 10-day AOT will not be entered without these compensatory measures in place, including the pre-staging of a BDB/FLEX AFW pump. For the BDB equipment, there are multiple pumps in each role available in the onsite BDB equipment storage building at SPS, and even more equivalent pumps available at North Anna Power Station that could be transported to SPS and deployed if needed. There are also procedural restrictions for having more than one of three BDB/FLEX pumps out of service at a time. Similarly, there are diverse and redundant FLEX qualified water sources available to provide AFW supply, including both units' Emergency Condensate Storage Tanks (ECSTs), the

> Emergency Condensate Makeup Tank (ECMT), and the BDB High-Capacity Pump, which can essentially draw water from the CW Discharge Canal indefinitely. Additional backup water sources are also available in the unlikely event they would be needed (e.g., Condensate Storage Tanks, Fire Protection Tanks) The high degree of redundancy of equipment and water sources and the existing procedural requirements will ensure availability is maintained for key SSCs relied upon in the compensatory measures and the Tier 2 evaluation.

The NRC staff concludes that the licensee has appropriately assessed the contribution to plant risk while the equipment covered by the proposed AOT change is out of service and the assessment to provide reasonable assurance that no risk significant configurations exist for very limited risk impact during the 10-day AOT of the requested change.

Tier 3: Risk-Informed Plant Configuration Control and Management

In its letter dated June 20, 2022, the licensee stated that the 10 CFR 50.65(a)(4) (i.e., Maintenance Rule) program at Surry performs PRA analyses of planned maintenance configurations in advance. The licensee further stated that configurations that approach or exceed the NUMARC 93-01 "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (ML110280084) risk limits are identified and either avoided or addressed by risk management actions. In addition, the configuration analysis and risk management processes are proceduralized in accordance with the requirements of 10 CFR 50.65(a)(4). The license also states:

> To support the SPS 10 CFR 50.65(a)(4) program, a dedicated PRA model is used to perform configuration risk analysis. The model uses the R06a model as a framework with some adjustments to optimize the model for configuration risk calculations. The model allows for quantitative Level 1 and Level 2 (LERF) assessments of internal events and internal flood hazards for at-power configurations. Risk during shutdown configurations and risks due to other hazards are assessed qualitatively. Changes in plant configuration or PRA model features are dispositioned and managed by Dominion Energy's PRA configuration control process. Procedures are in place to ensure actions are taken as necessary to qualitatively assess configurations outside the scope of the PRA model.

In its response to RAI-2 dated October 27, 2022, the licensee stated that the impacts due to seasonal variations to risk are accounted for in the Surry risk monitor model with indirect effects on grid instability, severe cold, plugging of circulating water traveling screens, and tornadoes.

Based on the above, the NRC staff finds that the licensee's risk-informed plant configuration and control program for compliance with 10 CFR 50.65(a)(4) provides reasonable assurance that the risk impact for out-of-service equipment is assessed and managed appropriately and is consistent with the guidance in RG 1.174, Revision 3, and 1.177, Revision 2, and is discussed further in Key Principle 5 below.

3.1.5 Key Principle 5 – NRC Staff Evaluation

RG 1.174, Revision 3, and RG 1.177, Revision 2, establish the need for an implementation and monitoring program to ensure that extensions to TS AOTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common-cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change.

An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. In its application dated June 20, 2022, the licensee stated in part, that:

> The impact of the proposed change will be monitored to ensure high levels of performance are maintained. The proposed AOT will limit individual instances of AFW cross-connect unavailability to 10 days when performing maintenance that would render the opposite unit's three AFW pumps inoperable and was determined to be reasonable and safe with the identified compensatory measures in place to mitigate risk. The Maintenance Rule, implemented in accordance with 10 CFR 50.65, will ensure that maintenance appropriately balances reliability and availability of the systems, structures and components (SSCs) related to the AFW cross-connect by tracking unavailability hours or by trending integrated risk.

These actions, in conjunction with the corrective action program, will ensure any adverse trends in AFW availability or equipment reliability performance are promptly and effectively addressed.

Key assumptions and sources of uncertainty were reviewed in preparation of the risk assessment to support the LAR. Dominion Energy Virginia documents assumptions and sources of uncertainty for the probabilistic risk assessment (PRA) in the SPS PRA notebook, SU-NOTEBK-PRA-SPS-QU.4, Revision 2. Sources of uncertainty were determined to be appropriately addressed for this application.

In its response to RAI-3 dated October 27, 2022, the licensee provided additional information regarding SSC performance monitoring. In its response, the licensee confirmed that the impact of the proposed AOT will limit individual instances of AFW pump cross-connect unavailability to 10 days when performing maintenance that would render the opposite unit's three AFW pumps inoperable. The licensee determined this to be reasonable and safe with the identified compensatory measures in place to mitigate risk. The Maintenance Rule program implemented in accordance with 10 CFR 50.65 will ensure that maintenance appropriately balances reliability and availability of the SSCs related to the AFW pump cross-connect by tracking unavailability hours or by trending integrated risk. The response further stated that the PRA’s Maintenance Rule input reflects the as-built plant and plant operating practices.

Based on the above, the NRC staff concludes that the implementation and monitoring program for the proposed LAR satisfies the Key Principle 5 of RG 1.174 and RG 1.177, because all affected SSCs are within the Maintenance Rule program, which is used to monitor changes to the reliability and availability of these SSCs.

Compensatory Measures

In Section 3.2 of its letter dated June 20, 2022, the licensee provided the following compensatory measures to be put in place to mitigate risk prior to entering the 10-day extended AOT:

- Additional AFW pump system maintenance, including associated water sources, or changes in plant configuration that would result in a risk significant configuration will be precluded;

- Weather conditions will be monitored and AFW pump maintenance affecting the operability of the opposite unit cross-connect will not be scheduled if severe weather conditions are anticipated;

- The steam-driven AFW pump will be controlled as “Protected Equipment.”

- The Technical Requirements Manual compensatory actions to address 10 CFR 50.65(a)(4) fire risk related to AFW pump cross-connect unavailability, which include periodic walkdowns in relevant fire areas, will be taken; and

- The BDB/FLEX AFW pump will be pre-staged to provide AFW defense-in-depth comparable to the AFW pump cross-connect.

In its response to RAI-4 dated October 27, 2022, the licensee stated:

> Several compensatory actions identified in this application rely on the availability of key components to backup safety functions if needed. The proposed 10-day AOT will not be entered without these compensatory measures in place, including the pre-staging of a BDB/FLEX AFW pump. For the BDB equipment, there are multiple pumps in each role available in the onsite BDB equipment storage building at SPS, and even more equivalent pumps available at North Anna Power Station that could be transported to SPS and deployed if needed. There are also procedural restrictions for having more than one of three BDB/FLEX pumps out of service at a time. Similarly, there are diverse and redundant FLEX qualified water sources available to provide AFW supply, including both units' Emergency Condensate Storage Tanks (ECSTs), the ECMT, and the BDB High-Capacity Pump, which can essentially draw water from the CW Discharge Canal indefinitely. Additional backup water sources are also available in the unlikely event they would be needed (e.g., Condensate Storage Tanks, Fire Protection Tanks) The high degree of redundancy of equipment and water sources and the existing procedural requirements will ensure availability is maintained for key SSCs relied upon in the compensatory measures and the Tier 2 evaluation.

The NRC staff reviewed the proposed compensatory measures with respect to preventing and mitigating the risk associated with certain initiating events including loss-of-offsite power or loss-of-normal heat sink to confirm these voluntary compensatory measures provide reasonable assurance that the licensee’s proposed actions were not overly relied upon to manage risk.

The NRC staff reviewed the information and concluded that the measures to protect equipment are appropriate and conservative with respect to the risk assessment in that the measures reduce the likelihood of maintenance affecting the availability of risk-important components. Based on the above, the NRC finds that the licensee has demonstrated reasonable assurance of adequate capability of design features without an overreliance on temporary or portable equipment to reduce risk. The licensee is providing FLEX pumps and diverse water sources to add defense-in-depth. Based on the above, the NRC staff concludes that reliance on these measures to reduce risk is appropriate and not inconsistent with the risk assessment modeling. The NRC staff finds that using compensatory actions are consistent with the DID philosophy of RG 1.174 and RG 1.177.

3.2 NRC Staff Conclusion

The NRC staff finds that the risk impact of the licensee's request as estimated by ∆CDF and ∆LERF, ICCDP, and ICLERP, is consistent with the acceptance guidelines specified in RG 1.177, Revision 2, RG 1.174, Revision 3, and the NRC staff guidance outlined in Chapters 19.1 and 16.1 of NUREG-0800. The licensee's methodology for assessing the risk impact is accomplished using PRA models of sufficient scope and technical adequacy based on a review of the model consistent with the guidance of RG 1.200, Revision 2. For external hazards which do not have PRA models, the licensee used bounding analyses. The NRC staff concludes that the licensee has followed the three-tiered approach and performance monitoring programs outlined in RG 1.177, Revision 2.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Commonwealth of Virginia’s State official was notified of the proposed issuance of the amendments on January 30, 2023. On January 30, 2023, the State official confirmed that the Commonwealth of Virginia had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration published in the *Federal Register* on September 6, 2022 (87 FR 54555) and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributors: Naeem Iqbal, NRR
Angelo Stubbs, NRR

Date: March 10, 2023

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 309 AND 309, RE: LICENSE AMENDMENT REQUEST REGARDING A 10-DAY ALLOWED OUTAGE TIME FOR OPPOSITE UNIT AUXILIARY FEEDWATER CROSS-CONNECT CAPABILITY (EPID: L-2022-LLA-0089) DATED MARCH 10, 2023

DISTRIBUTION:
Public
LPL2-1 R/F
RidsACRS_MailCTR Resource
RidsRgn2MailCenter Resource
RidsNrrDorlLpl2-1 Resource
RidsNrrPMSurry Resource
RidsNrrLAKGoldstein Resource
RidsNrrDssScpb Resource
RidsNrrDssStsb Resource
RidsNrrDraAplb Resource
CMoulton, NRR
NIqbal, NRR
AStubbs, NRR
JWilson, NRR

ADAMS Accession No.: ML23030B847

OFFICE	DORL/LPL2-1/PM	DORL/LPL2-1/LA	DRA/APLB/BC (A)	DSS/SCPB/BC
NAME	JKlos (SWilliams for)	KGoldstein	JRobinson	BWittick
DATE	1/31/2023	3/10/2023	12/15/2022	12/8/2022
OFFICE	DSS/STSB/BC	OGC - NLO	DORL/LPL2-1/BC	DORL/LPL2-1/PM
NAME	VCusumano	KDowling	MMarkley	JKlos
DATE	2/2/2023	2/22/2023	3/10/2023	3/10/2023

OFFICIAL RECORD COPY