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

July 26, 2023

Mr. David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 233 AND 233 RE: ADOPTION OF TSTF-577, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS," REVISION 1 (EPID L-2022-LLA-0115)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 233 to Renewed Facility Operating License No. NPF-72 and Amendment No. 233 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively. The amendments are in response to your letter dated August 10, 2022, as supplemented by letters dated December 7, 2022, and May 22, 2023. The amendments for other plants included in your letter dated August 10, 2022, will be issued under separate cover.

The amendment revises the "Steam Generator (SG) Program" and the "Steam Generator (SG) Tube Inspection Report" technical specifications based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections" (TSTF 577), and the associated NRC staff safety evaluation of TSTF-577.

A copy of the related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 233 to NPF-72
2. Amendment No. 233 to NPF-77
3. Safety Evaluation
4. Notices and Environmental Findings

cc: Listserv

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

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
Renewed License No. NPF-72

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

   A. The application for amendment by Constellation Energy Generation, LLC (the licensee) dated August 10, 2022, as supplemented by letters dated December 7, 2022, and May 22, 2023, 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 2.C.(2) of Renewed Facility Operating License No. NPF-72 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2023.07.26 13:39:06 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 26, 2023

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

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
Renewed License No. NPF-77

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

   A. The application for amendment by Constellation Energy Generation, LLC (the licensee) dated August 10, 2022, as supplemented by letters dated December 7, 2022, and May 22, 2023, 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 2.C.(2) of Renewed Facility Operating License No. NPF-77 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2023.07.26 13:39:26 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 26, 2023

ATTACHMENT TO LICENSE AMENDMENT NOS. 233 AND 233

RENEWED FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

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

Renewed Facility Operating Licenses

REMOVE	INSERT
License NPF-72	License NPF-72
-3-	-3-
License NPF-77	License NPF-77
-3-	-3-

Technical Specifications

REMOVE	INSERT
Page 5.5-7	Page 5.5-7
Page 5.5-8	Page 5.5-8
Page 5.5-9	Page 5.5-9
Page 5.5-10	Page 5.5-10
Page 5.5-11	Page 5.5-11
Page 5.5-12	Page 5.5-12
Page 5.5-13	Page 5.5-13
Page 5.5-14	Page 5.5-14
Page 5.5-15	Page 5.5-15
Page 5.5-16	Page 5.5-16
Page 5.5-17	Page 5.5-17
Page 5.5-18	Page 5.5-18
Page 5.5-19	Page 5.5-19
Page 5.5-20	Page 5.5-20
Page 5.5-21	Page 5.5-21
Page 5.5-22	Page 5.5-22
Page 5.5-23	Page 5.5-23
Page 5.5-24	Page 5.5-24
Page 5.6-6	Page 5.6-6
Page 5.6-7	Page 5.6-7

(2) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(2) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Constellation Energy Generation, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed a total of 1 gpm for all SGs.

3. The operational LEAKAGE performance criteria is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged. The following alternate tube plugging criteria may be applied as an alternative to the 40% depth based criteria:

For Unit 2, tubes with service-induced flaws located greater than 14.01 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 14.01 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. For Unit 2, portions of the tube below 14.01 inches from the top of the tubesheet are excluded from this requirement.

5.5.9 Steam Generator (SG) Program (continued)

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. For Unit 1, after the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

3. For Unit 2, after the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.

5.5.9 Steam Generator (SG) Program (continued)

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. For Unit 2, if crack indications are found in any SG tube from 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage, but may be deferred to the following refueling outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.3.

   If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR.

a. Demonstrate for each of the ESF filter systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Penetration
Control Room Ventilation (VC) Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 0.05%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the HEPA filter housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the HEPA filter housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 1%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	2.0%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is $<$ 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	$\geq$ 5400 cfm and $\leq$ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	$\geq$ 60,210 cfm and $\leq$ 73,590 cfm per train
FHB Ventilation System	$\geq$ 18,900 cfm and $\leq$ 23,100 cfm

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate for each of the ESF filter systems that a bypass test of the combined HEPA filters and damper leakage shows a total bypass specified below at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.12.4 and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 60,210 cfm and ≤ 73,590 cfm per train	≤ 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm	≤ 1%

f. Demonstrate that the heaters for each of the ESF filter systems dissipate the value specified below when tested in conformance with ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR.

ESF Ventilation System	Wattage
VC Filtration System	≤ 29.9 kW and ≥ 24.5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas system, the quantity of radioactivity contained in gas decay tanks or fed into the off gas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits, and

   3. a clear and bright appearance with proper color or a water and sediment content within limits;

b. Other properties of new fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e) as modified by approved exemptions.

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

5.5.16 Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

2. For each door, seal leakage rate is:

i. $< 0.0024\ L_a$, when pressurized to ≥ 3 psig, and
ii. $< 0.01\ L_a$, when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications,” or of the battery manufacturer of the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

5.5.18 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation (VC) Filtration System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

5.5.18 Control Room Envelope Habitability Program (continued)

a. The definition of the CRE and CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the VC Filtration System, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfilitered air inleakage limit for radiological challenges is the inleakage flow rate assumend in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The Provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.5.20 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1 and 2;

c. When a RICT is being used, any change to the plant configuration change, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

5.5.20 Risk Informed Completion Time Program (continued)

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods approved for use with this program, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

5.6 Reporting Requirements

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

   1. The nondestructive examination techniques utilized;

   2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

   3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

   4. The number of tubes plugged during the inspection outage.

5.6 Reporting Requirements

5.6.9 Steam Generator (SG) Tube Inspection Report (continued)

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

f. The results of any SG secondary side inspections;

g. For Unit 2, the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report;

h. For Unit 2, the calculated accident induced leakage rate from the portion of the tubes below 14.01 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.11 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

i. For Unit 2, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-72

AND

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-77

CONSTELLATION ENERGY GENERATION, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

Application	Safety Evaluation Date
• August 10, 2022, (ML22222A068) • December 7, 2022, (ML22342B230) • May 22, 2023, (ML23143A136)	July 26, 2023
	Principal Contributor to Safety Evaluation
	• Clinton Ashley

1.0 PROPOSED CHANGES

By letter dated August 10, 2022, as supplemented by letters dated December 7, 2022, and May 22, 2023, Constellation Energy Generation, LLC (the licensee) requested changes to the technical specifications (TSs) for Braidwood Station (Braidwood), Unit Nos. 1 and 2. In its letter dated August 10, 2022, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line-Item Improvement Process (CLIIP). The proposed changes would revise the “Steam Generator (SG) Program” and the “Steam Generator (SG) Tube Inspection Report” TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, “Revised Frequencies for Steam Generator Tube Inspections” (TSTF-577) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation (SE) of TSTF-577 (ML21098A188). This request was removed from the CLIIP process because of the complexity caused by not completing the required 100 percent SG tube inspections before the date of the request. This request was removed from the CLIIP process because of the complexity caused by not completing the required 100 percent SG tube inspections before the date of the request.

Enclosure 3

The tubes within an SG function as an integral part of the reactor coolant pressure boundary and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

Braidwood, Unit No. 1, SGs have Alloy 690 thermally treated (Alloy 690TT) tubes. Braidwood, Unit No. 2, SGs have Alloy 600 thermally treated (Alloy 600TT) tubes.

1.1 Proposed TS Changes to Adopt TSTF-577

In accordance with NRC staff-approved TSTF-577, the licensee proposed changes that would revise Braidwood TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.9, "Steam Generator (SG) Tube Inspection Report." Specifically, the licensee proposed the following changes to adopt TSTF-577:

TS 5.5.9, "Steam Generator (SG) Program":

- TS 5.5.9 introductory paragraph and paragraph b.1 would be revised by replacing "steam generator" with "SG" in a few instances.
- TS 5.5.9.d.2 and TS 5.5.9.d.3 would be revised by deleting the requirement to base inspection frequency on the more restrictive metric between either the effective full power months (EFPM) or refueling outage and to use just the EFPM metric.
- TS 5.5.9.d.2 and TS 5.5.9.d.3 would be revised by deleting the allowance to extend the inspection period by 3 EFPM and by deleting the discussion of prorating inspections.
- TS 5.5.9.d.2 would be revised by deleting the requirement to inspect 100 percent of the tubes during each period in paragraphs d.2.a, d.2.b, d.2.c, and d.2.d (144, 120, 96, and 72 EFPM, respectively) and by adding the requirement to inspect 100 percent of the tubes every 96 EFPM.
- TS 5.5.9.d.3 would be revised by changing the requirement to inspect 100 percent of the tubes at periods of 120, 96, and 72 EFPM to 54 EFPM. A 72 EFPM inspection period would be permitted if SG tubing has never experienced cracking (not including regions exempt from inspection by alternate repair criteria) and the SG inspection was performed with enhanced probes. A description of the enhanced probe inspection would be added.
- TS 5.5.9.d.4 would be revised by changing the next inspection after crack indications are found from "shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)" to "shall be at the next refueling outage," for both Unit 1 and Unit 2.
- TS 5.5.9.d.4 would be revised by adding an additional phrase for Unit 2 that permits deferring SG inspections after cracking indications are found if the 100 percent inspection was performed with enhanced probes.

TS 5.6.9, "Steam Generator (SG) Tube Inspection Report":

- Existing reporting requirement b. would be renumbered as c. and be revised by editorial and punctuation changes.

- New reporting requirement b. would be added to require the nondestructive examination (NDE) techniques utilized for tubes with increased degradation susceptibility be reported.

- Existing reporting requirement c. would be renumbered as c.1. and be revised by editorial and punctuation changes.

- Existing reporting requirement d. would be renumbered as c.2. and be revised to note that the location, orientation (if linear), measured size (if available), and voltage response do not need to be reported for tube wear indications at support structures that are less than 20 percent through-wall. However, the total number of tube wear indications at support structures that are less than 20 percent through-wall would be reported.

- New reporting requirement d. would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis methodology, inputs, and results.

- Existing reporting requirement e. would be renumbered as c.4. and be revised by editorial and punctuation changes.

- Existing reporting requirement f. would be renumbered as e. and be revised by editorial and punctuation changes.

- New reporting requirement f. would be added to require the results of any SG secondary side inspections be reported.

- Existing reporting requirement g. would be renumbered as c.3. and be revised to add the requirements to report a description of the condition monitoring assessment, the margin to the tube integrity performance criteria, and a comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. In addition, the requirement to report the results of tube pulls and in-situ testing would be deleted.

- Existing reporting requirements h., i., and j. would be renumbered as reporting requirements g., h., and i. and be revised by punctuation changes.

1.2 Additional Proposed TS Changes

In addition to the changes proposed consistent with the traveler discussed in Section 1.1, the licensee proposed the following variations in license amendment request (LAR) Section 2.2, “Variations.”

1.2.1 Editorial Variations

The licensee noted that Braidwood uses different numbering than the Standard Technical Specifications (STS) on which TSTS-577 was based. Specifically, Braidwood uses TS 5.6.9 for “Steam Generator (SG) Tube Inspection Report” and the TSTF-577 uses TS 5.6.7. In addition, Braidwood has common TS for Unit 1 and Unit 2, but each unit uses a different tubing alloy. As such, Braidwood TS 5.5.9 includes requirements specific to Unit 1 and requirements specific to Unit 2 that results in paragraph numbering differences when compared to TSTF-577.

1.2.2 Other Variations

The licensee noted that the Braidwood SG program TS currently contain a provision for an alternate tube plugging criteria. In its August 10, 2022, letter, the licensee specifies that the descriptions of the alternate tube plugging criteria in the proposed change are equivalent to the descriptions in the current Braidwood TSs.

The licensee also noted that the Braidwood TS contain two requirements that differ from the STS on which TSTF-577 was based. First, Braidwood TS 5.5.9.c currently states, in part: “[t]he following alternate tube plugging criteria *shall* be applied as an alternative to the 40% depth based criteria:”, whereas, TSTF-577 states: “[t]he following alternate tube plugging [or repair] criteria *may* be applied as an alternative to the 40% depth based criteria:” (emphasis added). The licensee proposed to change the word “shall” to “may” in Braidwood TS 5.5.9.c, to align with the wording in TSTF-577. Second, for TS 5.5.9.b.2, the accident induced leakage performance criterion states in part, “Leakage is not to exceed a total of 1 gpm [gallon per minute] for all SGs,” whereas TSTF-577 states in part, “Leakage is not to exceed [1 gpm] per SG.” The licensee proposed to retain the current TS leakage requirements.

2.0 REGULATORY EVALUATION

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.36(c)(5), “Administrative controls,” state that “[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4.” TS Section 5.0, “Administrative Controls,” requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG program, are listed in the administrative controls Section of the TS to operate the facility in a safe manner.

The NRC staff’s guidance for the review of TSs is in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Chapter 16.0, “Technical Specifications,” Revision 3, dated March 2010 (ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STS for each of the LWR nuclear designs. Accordingly, the NRC staff’s review includes consideration of whether the proposed changes are consistent with NUREG-1431[1], as modified by NRC-approved travelers.

TSTF-577 revised the STS related to SG tube inspections and SG tube inspection reporting requirements. The NRC approved TSTF-577, under the CLIIP on April 14, 2021 (ML21099A086).

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[1] U.S. Nuclear Regulatory Commission, “Standard Technical Specifications, Westinghouse Plants,” NUREG-1431, Volume 1, “Specifications,” and Volume 2, “Bases,” Revision 5, September 2021 (ML21259A155 and ML21259A159, respectively).

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes to Adopt TSTF-577

The NRC staff compared the licensee's proposed TS changes in Section 1.1 of this SE against the changes approved in TSTF-577. In accordance with SRP Chapter 16.0, the NRC staff determined that the STS changes approved in TSTF-577 are applicable because Braidwood, Unit Nos. 1 and 2, are pressurized-water reactors (PWRs) and the NRC staff approved the TSTF-577 changes for PWRs. The NRC staff finds that the licensee's proposed changes to the Braidwood, Unit Nos. 1 and 2, TSs in Section 1.1 of this SE are consistent with those found acceptable in TSTF-577.

In the SE of TSTF-577, the NRC staff concluded that the TSTF-577 changes to STS 5.5.9, "Steam Generator (SG) Program," and STS 5.6.7, "Steam Generator Tube Inspection Report," were acceptable because, as discussed in section 3.0 of that SE, they continued to ensure SG tube integrity and, therefore, protected the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion (explained in STS 5.5.9.b, items 1 and 2, respectively) will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). Additionally, the proposed changes to the reporting requirements will provide more detailed and consistent information to the NRC. Therefore, the NRC staff found that the proposed changes to the SG program and inspection reporting requirements were acceptable because they continued to meet the requirements of 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner. For these same reasons, the NRC staff concludes that the corresponding proposed changes to the Braidwood, Unit Nos. 1 and 2, TSs in section 1.1 of this SE continue to meet the requirements of 10 CFR 50.36(c)(5).

3.2 Additional Proposed TS Changes

3.2.1 Editorial Variations

The licensee noted that Braidwood uses different numbering than the STS on which TSTS-577 was based. Specifically, Braidwood uses TS 5.6.9 for "Steam Generator (SG) Tube Inspection Report" and the TSTF-577 uses TS 5.6.7. The NRC staff finds the different TS numbering is acceptable because it does not substantively alter TS requirements.

The licensee noted that Braidwood has common TSs for Unit 1 and Unit 2, but each unit uses a different tubing alloy. As such, Braidwood's current TS 5.5.9.d includes requirements specific to Unit 1 and requirements specific to Unit 2 that results in paragraph numbering differences when compared to TSTF-577. The NRC staff finds the proposed TS changes for Braidwood TS 5.5.9.d incorporated the intent of TSTF-577 changes while retaining the unit-specific format of Braidwood's current TS and appropriately addressed Unit differences in SG tube alloy, and therefore, are acceptable.

3.2.2 Other Variations

The licensee noted that the Braidwood SG program TS currently contains a provision for an alternate tube plugging criteria (TS 5.5.9.c). The NRC staff determined that the descriptions of the alternate tube plugging criteria in the proposed change are equivalent to the descriptions in the current Braidwood TS. The NRC staff finds the use of descriptions that are equivalent to the descriptions in the current Braidwood TS are acceptable because they do not substantively alter TS requirements or the applicability of TSTF-577.

The Braidwood TS contain two requirements that differ from the STS on which TSTF-577 was based. First, Braidwood TS 5.5.9.c currently states, in part: "[t]he following alternate tube plugging criteria *shall* be applied as an alternative to the 40% depth based criteria:", whereas TSTF-577 states: "[t]he following alternate tube plugging [or repair] criteria *may* be applied as an alternative to the 40% depth based criteria:" (emphasis added). The licensee proposed to change the word "shall" to "may" in Braidwood TS 5.5.9.c, to align with the wording in the STS on which TSTF-577 was based. The NRC staff finds the proposed change from "shall" to "may" in Braidwood TS 5.5.9.c aligns with the NRC staff-approved STS on which TSTF-577 was based, and therefore, is acceptable.

Second, for Braidwood TS 5.5.9.b.2, the accident induced leakage performance criterion states in part, "Leakage is not to exceed a total of 1 gpm for *all* SGs," whereas TSTF-577 states in part, "Leakage is not to exceed [1 gpm] *per* SG" (emphasis added). The NRC staff notes that the Braidwood TS leakage rate requirements are more restrictive than the STS and reflect NRC-approved changes contained in Amendment 144 to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood, Unit Nos. 1 and 2 (ML070810354), dated March 30, 2007. As part of the request to adopt TSTF-577, the licensee did not propose any changes to these leakage rate requirements. Therefore, based on the discussion above, the NRC staff considers the variation acceptable.

3.3 TS Change Consistency

The NRC staff reviewed the proposed TS changes for technical clarity and consistency with the existing requirements for customary terminology and formatting. The NRC staff finds that the proposed changes are consistent with Chapter 16.0 of the SRP and are therefore acceptable.

4.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

NOTICES AND ENVIRONMENTAL FINDINGS

RELATED TO

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-72

AND

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-77

CONSTELLATION ENERGY GENERATION, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

Application	Safety Evaluation Date July 26, 2023
• August 10, 2022, (ML22222A068) • December 7, 2022, (ML22342B230) • May 22, 2023, (ML23143A136)	Principal Contributor to Safety Evaluation Clinton Ashley

1.0 INTRODUCTION

By letter dated August 10, 2022, as supplemented by letters dated December 7, 2022, and May 22, 2023, Constellation Energy Generation, LLC (the licensee) requested changes to the technical specifications for Braidwood Station (Braidwood), Unit Nos. 1 and 2. In its letter dated August 10, 2022, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line-Item Improvement Process (CLIIP). The proposed changes would revise the “Steam Generator (SG) Program” and the “Steam Generator (SG) Tube Inspection Report” TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, “Revised Frequencies for Steam Generator Tube Inspections” (TSTF-577) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation of TSTF-577 (ML21098A188). This request was removed from the CLIIP process because of the complexity caused by not completing the required 100 percent SG tube inspections before the date of the request. This request was removed from the CLIIP process because of the complexity caused by not completing the required 100 percent SG tube inspections before the date of the request.

The supplements dated December 7, 2022, and May 22, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on October 4, 2022 (87 FR 60216).

2.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Illinois State official was notified of the proposed issuance of the amendment on June 6, 2023. The State official had no comments.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment relates, in part, to changes in recordkeeping, reporting, or administrative procedures or requirements. The amendment also relates, in part, to changing requirements with respect to the installation or use of facility components located within the restricted area as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 4, 2022 (87 FR 60216). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Date of issuance: July 26, 2023

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 233 AND 233 RE: ADOPTION OF TSTF-577, “REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS,” REVISION 1 (EPID L-2022-LLA-0115) DATED JULY 26, 2023

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ADAMS Accession No.: ML23188A129 **NRR-058**

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