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

February 17, 2023

Site Vice President
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT NO. 270 RE: ADOPTION OF TSTF-505, REVISION 2, “PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4b” (EPID L-2021-LLA-0014)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 270 to Renewed Facility Operating License (RFOL) No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the technical specifications (TSs) in response to your application dated February 8, 2021, as supplemented by letters dated April 8, 2021, May 16, 2022, August 19, 2022, and October 13, 2022.

The amendment revises the TS requirements to permit the use of risk-informed completion times in accordance with Technical Specifications Task Force (TSTF) Traveler (TSTF-505), Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b.”

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

Sincerely,

/RA/

Jason J. Drake, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 270 to NPF-38
2. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 270
Renewed License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated February 8, 2021, as supplemented by letters dated April 8, 2021, May 16, 2022, August 19, 2022, and October 13, 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 2.C.2 of Renewed Facility Operating License No. NPF-38 is hereby amended to read as follows:

   2. Technical Specifications and Environmental Protection Plan

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 270, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2023.02.17 11:32:30 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-38 and the Technical Specifications

Date of Issuance: February 17, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 270

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-38

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

Replace the following pages of Renewed Facility Operating License No. NPF-38 and the 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 License

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
3/4 3-4a	3/4 3-4a
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-18a	3/4 3-18a
3/4 3-18b	3/4 3-18b
3/4 5-3	3/4 5-3
3/4 6-9	3/4 6-9
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 6-18	3/4 6-18
3/4 6-19	3/4 6-19
3/4 6-20	3/4 6-20
3/4 7-4	3/4 7-4
3/4 7-9	3/4 7-9
3/4 7-9b	3/4 7-9b
3/4 7-11	3/4 7-11
3/4 7-12	3/4 7-12
3/4 7-43	3/4 7-43
3/4 8-1	3/4 8-1
3/4 8-2	3/4 8-2
3/4 8-2a	3/4 8-2a
3/4 8-9	3/4 8-9
3/4 8-14	3/4 8-14
6-10	6-10
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the NRC of any action by equity investors or successors in interest to Entergy Louisiana, LLC that may have an effect on 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

EOI is authorized to operate the facility at reactor core power levels not in excess of 3716 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 270, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Antitrust Conditions

(a) Entergy Louisiana, LLC shall comply with the antitrust license conditions in Appendix C to this renewed license.

(b) Entergy Louisiana, LLC is responsible and accountable for the actions of its agents to the extent said agent's actions contravene the antitrust license conditions in Appendix C to this renewed license.

Table 3.3-1 (Continued)

TABLE NOTATION

ACTION STATMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be documented by the On-Site Safety Review Committee in accordance with plant administrative procedures. The channel shall be returned to OPERABLE status prior to STARTUP following the next COLD SHUTDOWN.

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 500 psia.

(b) **An SIAS signal is first necessary to enable CSAS logic.**

* **The provisions of Specification 3.0.4 are not applicable.**

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be documented by the On-Site Safety Review Committee in accordance with plant administrative procedures. The channel shall be returned to OPERABLE status no later than prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Containment Pressure Circuit	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)

STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less that the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program or declare the associated value inoperable and take the ACTION required by Specification 3.7.1.5.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the tripped (D.C Relay energized) condition within 1 hour, the remaining Emergency Diesel Generator is OPERABLE, and the inoperable channel is restored to OPERABLE status within the next 48 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours. The surveillance requirements of Table 4.3-2 are waived for all channels while this action requirement is in effect.

ACTION 18 - With more than one channel inoperable, or if the inoperable channel cannot be placed in the trip (D.C. Relay energized) condition, declare the associated Emergency Diesel Generator inoperable and take the ACTION required by Specification 3.8.1.1. The surveillance requirements of Table 4.3-2 are waived for all channels while this action requirement is in effect.

ACTION 19 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue, provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour:

a. If the inoperable channel is to remain in the bypassed condition, the desirability of maintaining this channel in the bypassed condition shall be documented by the On-Site Safety Review Committee in accordance with plant administrative procedures. The channel shall be returned to OPERABLE status no later than prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN.

b. If the inoperable channel is required to be placed in the tripped condition, within 48 hours or in accordance with the Risk Informed Completion Time Program either restore the channel to OPERABLE status or place the channel in the bypassed condition. If the tripped channel can not be returned to OPERABLE status in 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours or place the tripped channel in bypass.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

TABLE 3.3-3 (Continued)

TABLE NOTATION

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
2.	Steam Generator Level	Steam Generator Level - Low Steam Generator ΔP (EFAS)

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
2.	Steam Generator Level	Steam Generator Level - Low Steam Generator ΔP (EFAS)

b. Restore at least one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Subsequent operation in the applicable MODE(S) may continue if one channel is restored to OPERABLE status and the provisions of ACTION 19 are satisfied.

3/4.5.2 ECCS SUBSYSTEMS - MODES 1, 2, AND 3

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection train,

b. One OPERABLE low-pressure safety injection train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*#.

ACTION:

a. With one ECCS subsystem inoperable due to one low pressure safety injection train inoperable, restore the inoperable train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than (a) and 100% of ECCS flow equivalent to a single OPERABLE ECCS subsystem available, restore the inoperable subsystem to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia and RCS average temperature to less than 500°F within the following 6 hours.

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*With pressurizer pressure greater than or equal to 1750 psia.
#With RCS average temperature greater than or equal to 500°F.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT VENTILATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve (CAP 103, CAP 104, CAP 203, and CAP 204) shall be OPERABLE and may be open at no greater than the 52° open position allowed by the mechanical stop for less than 90 hours per 365 days.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a containment purge supply and/or exhaust isolation valve(s) open for greater than or equal to 90 hours per 365 days at any open position, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one containment purge supply valve and/or exhaust isolation valve having a measured leakage rate exceeding the limits of Surveillance Requirement 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The cumulative time that the purge supply or exhaust isolation valves are open during the past 365 days shall be determined in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 Each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

4.6.1.7.3 Each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that the mechanical stops limit the valve opening to a position < 52° open.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWSP on a containment spray actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal. Each spray system flow path from the safety injection system sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With two containment spray systems inoperable, restore at least one spray system to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the water level in the containment spray header riser is > 149.5 feet MSL elevation.

b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is correctly positioned to take suction from the RWSP.

c. By verifying, that on recirculation flow, each pump develops a total head of greater than or equal to 219 psid when tested pursuant to the INSERVICE TESTING PROGRAM.

*With Reactor Coolant System pressure > 400 psia.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent trains of containment cooling shall be OPERABLE with one fan cooler to each train.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one train of containment cooling inoperable, restore the inoperable train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; restore the inoperable containment cooling train to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each train of containment cooling shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

   1. Starting each operational fan not already running from the control room and verifying that each operational fan operates for at least 15 minutes.

   2. Verifying a cooling water flow rate of greater than or equal to 625 gpm to each cooler.

b. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying that each fan starts automatically on an SIAS test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooler.

   3. Verifying that each cooling water control valve actuates to its full open position on a SIAS test signal.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the isolation valve inoperable for penetration(s) with closed system(s) either:

a. Restore the inoperable valve to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or

b. Isolate each affected penetration within 72 hours or in accordance with the Risk Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position and verify the affected penetration flow path is isolated once per 31 days following isolation, or

c. Isolate each affected penetration within 72 hours or in accordance with the Risk Informed Completion Time Program by use of at least one closed manual valve or blind flange and verify the affected penetration flow path is isolated once per 31 days following isolation, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Note: Isolation devices in a high radiation area may be verified by use of administrative means.

For all other penetrations, with one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

e. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or

f. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position and verify the affected penetration flow path is isolated once per 31 days following isolation, or

g. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one closed manual valve or blind flange and verify the affected penetration flow path is isolated once per 31 days following isolation, or

h. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 do not apply.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

b. Verifying that on a containment Radiation-High test signal, each containment purge valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Three emergency feedwater (EFW) pumps and two flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one steam supply to the turbine-driven EFW pump inoperable, restore the steam supply to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one steam supply to the turbine-driven EFW pump and one motor-driven EFW pump inoperable and the EFW flow paths able to deliver at least 100% flow to their respective steam generators, restore the steam supply or motor-driven EFW pump to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one steam supply to the turbine-driven EFW pump and both motor-driven EFW pumps inoperable and the EFW flow paths able to deliver at least 100% flow to their respective steam generators, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With the EFW system inoperable for reasons other than those described in ACTION (a), (b), or (c), and able to deliver at least 100% flow to either steam generator, restore the EFW system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With the EFW system inoperable for reasons other than those described in ACTION (a), (b), or (c), and able to deliver at least 100% combined flow to the steam generators, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With the EFW system inoperable and unable to deliver at least 100% combined flow to the steam generators, immediately initiate action to restore the ability to deliver at least 100% combined flow to the steam generators. LCO 3.0.3 and all other LCO ACTIONs requiring MODE changes are suspended until the EFW system is capable of delivering at least 100% combined flow to the steam generators.

g. Only as allowed by Surveillance Requirements 4.7.1.2(b) and 4.7.1.2(c), the provisions of Specifications 3.0.4 and 4.0.4 are not applicable to the turbine-driven EFW pump for entry into Mode 3.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES (MSIVs)

LIMITING CONDITION FOR OPERATION

3.7.1.5 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1, and
MODES 2, 3, and 4, except when all MSIVs are closed and deactivated.

ACTION:

MODE 1

With one MSIV inoperable, restore the valve to OPERABLE status within 8 hours or in accordance with the Risk Informed Completion Time Program or be in STARTUP within the next 6 hours.

MODES 2, 3 and 4

With one MSIV inoperable, close the valve within 8 hours and verify the valve is closed once per 7 days. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Note: Required to be performed for entry into MODES 1 and 2 only.

4.7.1.5 Each MSIV shall be demonstrated OPERABLE:

a. By verifying full closure within 8.0 seconds when tested pursuant to the INSERVICE TESTING PROGRAM.

b. By verifying each MSIV actuates to the isolation position on an actual or simulated actuation signal in accordance with the Surveillance Frequency Control Program.

3/4.7 PLANT SYSTEMS

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each Atmospheric Dump Valve (ADV) shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With the automatic actuation channel for one ADV inoperable, restore the inoperable ADV to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or reduce power to less than or equal to 70% RATED THERMAL POWER within the next 6 hours.

b. With the automatic actuation channels for both ADVs inoperable, restore one ADV to OPERABLE status within 1 hour or reduce power to less than or equal to 70% RATED THERMAL POWER within the next 6 hours.

c. With one ADV inoperable, for reasons other than above, restore the ADV to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The provisions of Specification 3.0.4 are not applicable provided one ADV is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.7.1.7 The ADVs shall be demonstrated OPERABLE:

a. By performing a CHANNEL CHECK in accordance with the Surveillance Frequency Control Program when the automatic actuation channels are required to be OPERABLE.

b. By verifying each ADV automatic actuation channel is in automatic with a setpoint of less than or equal to 1040 psia in accordance with the Surveillance Frequency Control Program when the automatic actuation channels are required to be OPERABLE.

c. By verifying one complete cycle of each ADV when tested pursuant to the INSERVICE TESTING PROGRAM.

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* ADV automatic actuation channels (one per ADV, in automatic with a setpoint of less than or equal to 1040 psia) are not required to be OPERABLE when less than or equal to 70% RATED THERMAL POWER for greater than 6 hours.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water and associated auxiliary component cooling water trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water and associated auxiliary component cooling water train OPERABLE, restore at least two trains to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 Each component cooling water and associated auxiliary component cooling water train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on SIAS and CSAS test signals.

c. In accordance with the Surveillance Frequency Control Program by verifying that each component cooling water and associated auxiliary component cooling water pump starts automatically on an SIAS test signal.

PLANT SYSTEMS

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 Two independent trains of ultimate heat sink (UHS) cooling towers shall be OPERABLE with each train consisting of a dry cooling tower (DCT) and a wet mechanical draft cooling (WCT) and its associated water basin with:

a. A minimum water level in each wet tower basin of 97% (-9.77 ft MSL)

b. An average basin water temperature of less than or equal to 89°F.

c. Fans as required by Table 3.7-3.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With 1 UHS train Inoperable, restore the Inoperable train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both UHS trains inoperable, restore at least one UHS train to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

PLANT SYSTEMS

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.12 Two independent essential services chilled water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With only one essential services chilled water loop OPERABLE, restore two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Each of the above required essential services chilled water loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the water outlet temperature is ≤ 42°F at a flow rate of ≥ 500 gpm.

c. Deleted

d. In accordance with the Surveillance Frequency Control Program, by verifying that each essential services chilled water pump and compressor starts automatically on a safety injection actuation test signal.

4.7.12.2 The backup essential services chilled water pump and chiller shall be demonstrated OPERABLE in accordance with Specification 4.7.12.1 whenever it is functioning as part of one of the required essential services chilled water loops.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

 1. Diesel oil feed tanks containing a minimum one hour supply of fuel, and

 2. A separate diesel generator fuel oil storage tank, and

 3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1a inoperable, demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter. Restore the offsite A.C. circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1b inoperable:

 (1) Demonstrate the OPERABILITY of the remaining A.C. circuits by performing Surveillance Requirements 4.8.1.1.1a (separately for each offsite A.C. circuit) within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator (unless it has been successfully tested in the last 24 hours) by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated.

 (2) Restore the diesel generator to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

ACTION: (Continued)

(a) The requirement for restoration to OPERABLE status within 72 hours may be extended to 10 days if a temporary emergency diesel generator is verified available, and

(b) If at any time the temporary emergency diesel generator availability cannot be met, either restore the temporary emergency diesel generator to available status within 72 hours (not to exceed 10 days from the time the permanent plant EDG originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one offsite A.C. circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; and, if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours (unless it is already operating) unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite A.C. circuit or diesel generator) to OPERABLE status in accordance with the provisions of ACTION statement a or b, as appropriate, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2a.4 performed under this ACTION statement satisfies the diesel generator test requirement of ACTION statement a or b.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

(1) All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

(2) When in MODE 1, 2, or 3, the steam-driven emergency feed pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

e. With two of the above required offsite A.C. circuits inoperable, restore one of the inoperable offsite A.C, circuits to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite A.C. circuit, follow ACTION statement a with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2a.4 performed under this ACTION statement satisfies the diesel generator test requirement of ACTION statement a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator, follow ACTION statement b with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable diesel generator.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank No. 3A-S and one associated full capacity charger (3A1-S or 3A2-S).

b. 125-volt Battery Bank No. 3B-S and one associated full capacity charger (3B1-S or 3B2-S).

c. 125-volt battery Bank No. 3AB-S and one associated full capacity charger (3AB1-S or 3AB2-S).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and at least one associated charger shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 125 volts on float charge.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With one of the required divisions of A.C. ESF busses not fully energized, reenergize the division within 8 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. SUPS bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. SUPS bus within 2 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. SUPS bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not connected to its associated battery bank, reconnect the D.C. bus from its associated OPERABLE battery bank within 2 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

6.5.19 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 MODES 1 and 2;

c. When a RICT is being used, any change to the plant configuration, 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.

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.

Pages 6-12
through
page 6-13
not used

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 270 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated February 8, 2021 (Reference 1), as supplemented by letters dated April 8, 2021 (Reference 2), May 16, 2022 (Reference 3), August 19, 2022 (Reference 4), and October 13, 2022 (Reference 5), Entergy Operations, Inc (Entergy, the licensee) submitted a license amendment request (LAR) for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b," dated July 2, 2018 (TSTF-505) (Reference 6). The U.S. Nuclear Regulatory Commission (NRC, the Commission) issued a final revised model safety evaluation (SE) to be used when preparing a plant-specific SE of an LAR to adopt TSTF-505, on November 21, 2018 (Reference 7).

The licensee has proposed variations from the TS changes approved in TSTF-505, which are provided in section 2.3 of the LAR and evaluated in section 3.0 of this SE.

The NRC staff participated in a regulatory audit in October 2021. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), as needed. On October 7, 2021, the NRC staff issued an audit plan (Reference 8).

The supplemental letters dated May 16, 2022, August 19, 2022, and October 13, 2022, 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 May 18, 2021 (86 FR 26954).

2.0 REGULATORY EVALUATION

Title 10 of the *Federal Code of Regulations* (10 CFR) Part 50 provides the general provisions for "Domestic Licensing of Production and Utilization Facilities." The general provisions include but are not limited to establishing the regulatory requirements that a licensee must adhere to for the submittal of a license application. The NRC staff has identified the following applicable sections within 10 CFR Part 50 for the staff's review of a licensee's application to adopt TSTF-505.

- 10 CFR 50.36, "Technical Specifications," paragraphs (c)(2), "Limiting conditions for operation," and (c)(5), "Administrative controls"

- 10 CFR 50.55a, "Codes and standards," paragraph (h), "Protection and safety systems"

- 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule)

NRC Regulatory Guides (RGs) provide one way to ensure that the codified regulations continue to be met. The NRC staff considered the following guidance, along with industry guidance endorsed by the NRC, during its review of the proposed changes:

- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 9) and RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated December 2020 (Reference 10).

- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (Reference 11).

- RG 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated January 2021 (Reference 12).

- NRC Regulation (NUREG)-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs [Probabilistic Risk Assessments] in Risk-Informed Decisionmaking," dated March 2017 (Reference 13).

- NUREG-1432, ""Standard Technical Specifications Combustion Engineering Plants" [STS], "Volume 1, "Specifications," and Volume 2, "Bases," dated September 2021 (Reference 14).

- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 16.1, "Risk-Informed Decision Making: Technical Specifications," dated March 2007 (Reference 15).

The licensee's submittals cite RG 1.200, Revision 2 as applicable guidance. The updates in RG 1.200, Revision 3 do not include any technical changes that would impact the consistency with Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" dated October 2012 (Reference 16). The NRC staff issued a final model SE approving NEI 06-09 with limitations and conditions on May 17, 2007 (Reference 17). Therefore the NRC staff finds the RG 1.200, Revision 3 is also applicable for use in the adoption of TSTF-505.

2.1 Description of Risk-Informed Completion Time Program

The TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee must shut down the reactor or follow any remedial or required action (e.g., testing, maintenance, or repair activity) permitted by the TSs until the condition can be met. The remedial actions (i.e., ACTIONS) associated with an LCO contain conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s) (CT). The CTs are referred to as the “front stops” in the context of this SE. For certain conditions, the TSs require exiting the Mode of Applicability of an LCO (i.e., shut down the reactor).

The licensee’s submittal requested approval to add a RICT program to the Administrative Controls Section of the TS, and to modify selected CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. Consistent with the section on NUREG-1432, “Standard Technical Specifications Combustion Engineering Plants” (STS), in table 1, “Conditions Requiring Additional Technical Justification,” of TSTF-505 for “Conditions Requiring Additional Technical Justification” (Specifications),” the licensee provided several plant-specific LCOs and associated Actions proposed to be included in the RICT program, along with additional justification. NRC staff review of these variations and the justification is provided in section 3.0 of this SE.

The licensee is proposing no changes to the design of the plant or any operating parameter, and no new changes to the design-basis in the proposed changes to the TSs. The effect of the proposed changes when implemented will allow CTs to vary, based on the risk significance of the given plant configuration (i.e., the equipment out of service at any given time), provided that the system(s) retain(s) the capability to perform the applicable safety function(s) without any further failures (e.g., one train of a two-train system is inoperable). These restrictions on inoperability of all required trains of a system ensure that consistency with the defense-in-depth (DID) philosophy is maintained by following existing guidance when the capability to perform TS safety function(s) is lost.

The proposed RICT program uses plant-specific operating experience for component reliability and availability data. Thus, the allowances permitted by the RICT program are directly reflective of actual component performance in conjunction with component risk significance.

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

The NRC staff reviewed the licensee’s PRA peer review history and results, alternative methods, and proposed approaches to determine if they are technically acceptable for use in the proposed RICT extensions. The NRC staff also reviewed the licensee’s proposed RICT program to determine if it provides the necessary administrative controls to permit CT extensions for consistency with NEI 06-09-A.

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed licensing basis

(LB) changes meet the five key principles provided in RGs 1.174 and 1.177 and the three-tiered approach outlined in 1.177.

Key Principle 1: Evaluation of Compliance with Current Regulations

Paragraph 50.36(c)(2) of 10 CFR states, in part, that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility and that 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 LCO can be met.

The CTs in the current TSs were established using experiential data, risk insights, and engineering judgement. The RICT program provides the necessary administrative controls to permit extension of CTs and, thereby, delay reactor shutdown or Required Actions, if risk is assessed and managed appropriately within specified limits and programmatic requirements and the safety margins and DID remains sufficient. The option to determine the extended CT in accordance with the RICT program allows the licensee to perform an integrated evaluation in accordance with the methodology prescribed in NEI 06-09-A and proposed TS 6.5.19, "Risk Informed Completion Time Program." The RICT is limited to a maximum of 30 days (termed the "back stop").

The typical CT is modified by the application of the RICT program as shown in the following example. The changed portion is indicated in italics.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days *OR* *In accordance with the Risk-Informed Completion Time Program*

In attachment 2, "Proposed Technical Specification Changes (Mark-up)," and enclosure 1, "List of Revised Required Actions to Corresponding PRA Functions," to the LAR, as supplemented, the licensee provided a list of the TSs, associated LCOs, and Required Actions for the CTs that included modifications and variations from the approved TSTF-505. The modifications and variations consisted of proposed changes to the Required Actions and CTs. Furthermore, consistent with table 1 of TSTF-505 for Waterford 3 TSs 3.4.3.1, 3.6.1.3, 3.7.1.6, 3.5.2, 3.6.2.1, and 3.6.2.2 in section 2.3 in attachment 1, "Description and Assessment of the Proposed Change," to the LAR, as supplemented, the licensee included additional technical justification to demonstrate the acceptability for including these TSs in the RICT program. The NRC staff reviewed the proposed changes to the TSs, associated LCOs, Required Actions and CTs provided by the licensee for the scope of the RICT program and concluded that, with the incorporation of the RICT program, the required performance levels of equipment specified in LCOs are not changed, only the required CT for the Required Actions are modified, such that 10 CFR 50.36(c)(2) will continue to be met. Based on the discussion provided above, the NRC staff finds that the RICT program provided in section 2.0 of this SE, LCOs, Required Actions, and CTs meet the first key principle of RG 1.174 and RG 1.177.

Although the following TS Actions proposed for a RICT provide for redundancy of function, the NRC staff determined that they are either identified for further evaluation within TSTF-505 or are outside the general scope of TSTF-505:

- LCO 3.6.2.1, Action a. for one containment spray system inoperable.

- LCO 3.6.2.2, Action for one train of containment cooling inoperable.

- LCO 3.7.1.2, Action d. for an inoperable emergency feedwater (EFW) system that remains capable of delivering at least 100 percent flow to either steam generator (SG).

- LCO 3.7.1.5, Action for one inoperable main steam line isolation valve (MSIV).

- LCO 3.7.1.7, Action a. for an inoperable atmospheric dump valve (ADV) automatic actuation channel.

The above TS Actions are, therefore, evaluated further in section 3.3, "Conditions Requiring Additional Technical Justification," of this SE, to ensure that the function will be maintained.

<u>Key Principle 2: Evaluation of DID</u>

In RG 1.174, the NRC identified the following considerations used for evaluation of how the licensing basis change is maintained for the DID philosophy:

- 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 licensee requested to use the RICT program to extend the existing CTs for the respective TS LCOs prescribed in attachment 2 to the LAR, as supplemented. For the TS LCOs, in attachment 2 and enclosure 1 to the LAR, as supplemented, the licensee provided a description and assessment of the redundancy and diversity for the proposed changes. The NRC staff's evaluation of the proposed changes for these LCOs assessed Waterford 3's redundant or diverse means to mitigate accidents to ensure consistency with the plant licensing basis requirements using the guidance prescribed in RGs 1.174 and 1.177 and TSTF-505, to ensure adequate DID (for each of the functions) to operate the facility in the proposed manner (i.e., that the changes are consistent with the DID criteria).

Enclosure 1 to the LAR, as supplemented, provided information supporting the Waterford 3 evaluation of the redundancy, diversity, and DID for each TS LCO and TS Required Action as it relates to instrumentation and control (I&C), and electrical power systems. The NRC confirmed that the following TS LCOs are consistent with the DID philosophy:

- TS 3.3.1, "Reactor Protective Instrumentation"
- TS 3.3.2, "Engineered Safety Features Actuation System Instrumentation"
- TS 3.8.1.1, "A.C. [Alternating Current] Sources – Operating"
- TS 3.8.2.1, "DC [Direct Current] Sources – Operating"
- TS 3.8.3.1, "Onsite Power Distribution Systems – Operating")

For the TS LCOs specific to I&C (TSs 3.3.1 and 3.3.2), the NRC staff reviewed the specific trip logic arrangements, redundancy, backup systems, manual actions, and diverse trips specified for each of the protective safety functions and associated instrumentation as described in the associated Updated Final Safety Analysis Report (UFSAR) (Reference 18) sections, and as reflected in enclosure 1 to the LAR, as supplemented, for each I&C LCO above. The NRC staff verified, that in accordance with the Waterford 3 UFSAR and equipment and actions credited in enclosure 1 to the LAR, as supplemented, in all applicable operating modes, the affected protective feature would perform its intended function by ensuring the ability to detect and mitigate the associated event or accident when the CT of a channel is extended. Furthermore, the NRC staff concludes that there is sufficient I&C redundancy, diversity, and DID to protect against CCFs and potential single failure for the Waterford 3 instrumentation systems evaluated in enclosure 1 to the LAR during a RICT. There is at least one diverse means specified by the licensee for initiating mitigating action for each accident event, thus providing DID against a failure of instrumentation during the RICT for each TS LCO. The DID specified by the licensee does not overly rely on manual actions as the diverse means; therefore, there is not over-reliance of programmatic activities as compensatory measures. Therefore, the NRC staff finds that the intent of the plant's design criteria (e.g., safety functions) for the above TS LCOs related to I&C are maintained.

For the TS LCOs specific to electrical power systems, the Waterford 3 UFSAR states that the plant is designed such that the safety functions are maintained assuming a single failure within the electrical power system. Single failure requirements are typically suspended for the time that a plant is not meeting an LCO (i.e., in an ACTION statement). The NRC staff reviewed the information the licensee provided in the LAR, as supplemented, for the proposed TS LCOs and TS Bases, and the UFSAR to verify the capability of the affected electrical power systems to perform their safety functions (assuming no additional failures) is maintained. The staff verified that the design success criteria for the affected TS LCO stated in table E1-1, "List of Revised Required Actions to Corresponding PRA Functions," of enclosure 1 to the LAR supplement dated May 16, 2022, reflect the redundant or absolute minimum electrical power source/subsystem required to be operable to support the safety functions necessary to mitigate postulated design-basis accidents (DBAs), safely shutdown the reactor, and maintain the reactor in a safe shutdown condition. In addition, the NRC staff reviewed the risk management action (RMA) examples which provide reasonable assurance that the appropriate RMAs will be implemented to monitor and control risk. The NRC staff finds that the intent of the plant's design criteria (e.g., safety functions) applicable to the electrical power related TS LCOs provided above are maintained.

The NRC staff notes that while in a TS LCO condition, the redundancy of the function will be temporarily relaxed and, consequently, the system reliability will be degraded accordingly. The NRC staff examined the design information from the Waterford 3 UFSAR, and the risk-informed TS LCO conditions for the affected safety functions. Based on this information, the NRC staff confirmed that under any given DBA evaluated in the Waterford 3 UFSAR, the affected protective features maintain adequate DID.

Considering that the CT extensions will be implemented in accordance with the guidance in NEI 06-09-A that also considers RMAs, and the redundancy of the offsite and onsite power system, the NRC staff finds that the plant will maintain adequate DID. Therefore, the NRC staff finds the electrical power system related TS LCOs proposed by the licensee in enclosure 1 to the LAR, as supplemented, are acceptable for the RICT program.

The NRC staff reviewed all TS LCOs proposed by the licensee in enclosure 1 to the LAR, as supplemented, and concludes that the proposed changes do not alter the ways in which the Waterford 3 systems fail, do not introduce new CCF modes, and the system independence is maintained. The NRC staff finds that some proposed changes reduce the level of redundancy of the affected systems, and this reduction may reduce the level of defense against some CCFs; however, such reductions in redundancy and defense against CCFs are acceptable due to existing diverse means available to maintain adequate DID against a potential single failure during a RICT. The NRC staff finds that extending the selected CTs with the RICT program following loss of redundancy, but maintaining the capability of the system to perform its safety function, is an acceptable reduction in DID during the proposed RICT period provided that the licensee identifies and implements compensatory measures in accordance with the RICT program during the extended CT.

Based on the above, the NRC staff finds that the licensee's proposed changes are consistent with the NRC-endorsed guidance prescribed in NEI 06-09-A and satisfy the second key principle in RGs 1.174 and 1.177. Additionally, the NRC staff concludes that the changes are consistent with the DID philosophy as described in RG 1.174.

<u>Key Principle 3: Evaluation of Safety Margins</u>

Paragraph 50.55a(h) of 10 CFR requires, in part, that protection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph. Section 2.2.2, "Technical Specification Change Maintains Sufficient Safety Margin (Principle 3)," of RG 1.177 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 final safety analysis report are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties….

The licensee is not proposing in its LAR to change any quality standard, material, or operating specification. In the LAR, the licensee proposed to add a new program, the RICT Program, in section 5.0, “Administrative Controls,” of the Waterford 3 TSs, which would require adherence to NEI 06-09-A. NEI 06-09-A, Condition 2 in part, stipulates for the TS LCOs and action requirements to which the RMTS will apply, the LAR will provide justification with comparison of the TS functions to the PRA modeled functions of the structures, systems, or components (SSCs) subject to those LCO actions or an appropriate disposition or programmatic restriction will be provided.

The NRC staff evaluated the effect on safety margins when the RICT program is applied to extend the CT up to a backstop of 30 days in a TS condition with sufficient trains remaining operable to fulfill the TS safety function. Although the licensee will be able to have design-basis equipment out of service longer than the current TS allows, any increase in unavailability is expected to be insignificant and is addressed by the consideration of the single failure criterion in the design-basis analyses. Acceptance criteria for operability of equipment are not changed and, if sufficient trains remain operable to fulfill the TS safety function, the operability of the remaining train(s) ensures that the current safety margins are maintained. The NRC staff finds that if the specified TS safety function remains operable, sufficient safety margins would be maintained during the extended CT of the RICT program.

Safety margins are also maintained if PRA functionality is determined for the inoperable train, which would result in an increased CT. Credit for PRA functionality, as described in NEI 06-09-A, is limited to the inoperable train, loss-of-offsite power (LOOP), or component.

Based on the above, the NRC staff finds that the design basis analyses for Waterford 3 remain applicable and unchanged, that sufficient safety margins would be maintained during the extended CT, and that the proposed changes to the TSs do not include any change in the standards applied or the safety analysis acceptance criteria. The NRC staff concludes that the proposed changes meet 10 CFR 50.55a(h) and, therefore, Key Principle 3.

Key Principle 4: Change in Risk Consistent with the Safety Goal Policy Statement

NEI 06-09-A provides a methodology for a licensee to evaluate and manage the risk impact of extensions to TS CTs. Permanent changes to the fixed TS CTs are typically evaluated by using the three-tiered approach described in SRP section 16.1 and RG 1.177. This approach addresses the calculated change in risk as measured by the change in core damage frequency (CDF) and large early release frequency (LERF), as well as the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP); the use of compensatory measures to reduce risk; and the implementation of a configuration risk management program (CRMP) to identify risk-significant plant configurations.

The NRC staff evaluated the licensee’s processes and methodologies for determining that the change in risk from implementation of RICTs will be small and consistent with the intent of the Commission’s Safety Goal Policy Statement. In addition, the NRC staff evaluated the licensee’s proposed changes against the three-tiered approach in RG 1.177 for the licensee’s evaluation of the risk associated with a proposed TS CT change. The results of the NRC staff’s review are discussed below.

Tier 1: PRA Capability and Insights

Tier 1 evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) scope and acceptability of the PRA models and their application to the proposed changes and (2) a review of the PRA results and insights described in the licensee's application.

In enclosures 2, "Information Supporting Consistency with Regulatory Guide 1.200, Revision 2," and enclosure 4, "Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models," to the LAR and enclosure 1, "Response to Request for Additional Information," to the LAR supplement dated May 16, 2022, the licensee identified the following modeled hazards and alternate methodologies that the licensee used to assess the risk contribution for extending the CT of a TS LCO in the proposed Waterford 3 RICT program.

- Internal Events PRA (IEPRA) model (includes internal floods)

- Internal Fire PRA (FPRA) model

- Seismic Hazard: a CDF penalty of 4.24E-06 per year, and a LERF penalty of 1.94E-06 per year

- Other External Hazards: screened out from RICT program based on appendix 6-A of the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard ASME/ANS RA-Sa-2009 Addenda to ASME RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME/ANS RA-SB-2005 PRA standard) (Reference 19)

<u>Evaluation of Modeled PRAs</u>

In attachment 4, "Revisions to Information Supporting Consistency with Regulatory Guide 1.200, Revision 2," to enclosure 1 to the LAR supplement dated May 16, 2022, the licensee confirmed that the PRA models have been peer reviewed. The internal events PRA was peer reviewed using the ASME/ANS RA-Sb-2005 PRA standard as endorsed by RG 1.200, Revision 2 Further the IEPRA and FPRA were peer reviewed using the ASME/ANS RA-Sa-2009 PRA standard. This included for the IEPRA focused-scope peer reviews on internal flooding, LERF, and human reliability analysis. The licensee stated that it conducted an independent assessment process for closure of the facts and observations (F&Os) resulting from these peer reviews. The NRC staff confirmed that the licensee performed closure of the F&Os consistent with Appendix X to NEI 05-04, 07-12, and 12-13, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations" (Reference 20), as endorsed in RG 1.200, Revision 3. The NRC evaluated the remaining open F&Os, along with their dispositions. In the LAR supplement dated August 19, 2022, the licensee proposed a license condition to resolve the open human reliability F&Os using the NRC accepted process, which is described in section 3.0 of this SE. The portions of RG 1.200, Revisions 2 and 3 discussed above do not include any technical changes that would impact the consistency with NEI 06-09-A; therefore, the NRC staff finds these portions of the RG also applicable for use in the licensee's adoption of TSTF-505.

During the approval process for the Waterford 3 LAR to adopt the 10 CFR 50.69 categorization process, the NRC staff became aware that the Waterford 3 FPRA had not yet incorporated the updated ignition frequencies provided in NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009" (Reference 21). In response to NRC PRA Licensing Branch A (APLA) RAI 08.a and APLA RAI 09, the licensee performed sensitivity studies to assess the impact of the fire ignition frequencies on the RICT program. The results of the sensitivity studies are found in updated table E1-2, "In Scope TS/LCO Conditions RICT Estimate," in attachment 2 of enclosure 1 to the LAR supplement dated May 16, 2022, which demonstrate a significant impact on several RICT estimates. Therefore, the licensee provided, in enclosure 5, "List of Regulatory Commitments," to the LAR supplement dated May 16, 2022, a commitment to incorporate the NUREG-2169 ignition frequencies prior to RICT program implementation.

During the approval process for the Waterford 3 LAR to adopt the 10 CFR 50.69 categorization process, the NRC staff became aware that the Waterford 3 PRA models incorporate flexible and diverse coping (FLEX) strategies and equipment. The NRC staff noted in the NRC memorandum, "Assessment of the NEI 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis," dated May 30, 2017 (Reference 22), certain uncertainties related to FLEX modeling associated with failure rates of portable equipment, and that these uncertainties should be considered in the PRA models to stay consistent with the ASME/ANS RA-Sa-2009 PRA standard, . In response to APLA RAI 02 in the LAR supplement dated May 16, 2022, the licensee stated that sensitivity studies were performed on the proposed RICT TS LCOs with no FLEX credit that demonstrated minimal impact on RICT values.

The licensee considered PRA modeling uncertainties and their potential impact on the RICT program and identified, as necessary, the applicable RMAs to limit the impact of these uncertainties. In response to APLA RAI 08.b and c in the LAR supplement dated May 16, 2022, the licensee discussed the identification of key assumptions and sources of uncertainty along with providing the dispositions for impact on the risk-informed application or applicable sensitivities. In enclosure 2 to this supplement the licensee evaluated the Waterford 3 PRA models to identify the key assumptions and sources of uncertainty for this application, consistent with the RG 1.200, Revision 2, definitions, and using sensitivity and importance analyses to place bounds on uncertain processes, identify alternate modeling strategies, and provide information to users of the PRA.

The NRC staff finds that the licensee performed an adequate assessment to identify the potential sources of uncertainty, and that the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855 and associated Electric Power Research Institute (EPRI) TRs 1016737, "Treatment of Parameter and Model Uncertainity for Probabilistic Risk Assessments," December 2008 (Reference 23) and 1026511, "Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-informed Applications with a Focus on the Treatment of Uncertainty," December 2012 (Reference 24). Therefore, the NRC staff finds that the licensee has satisfied the guidance in RGs 1.174 and 1.177 and that the identification of assumptions and treatment of model uncertainties for risk evaluation of extended CTs is appropriate for this application and is consistent with the guidance in NEI 06-09-A.

The NRC staff reviewed the peer review history of the PRA models provided by the licensee in attachment 4 of enclosure 1 to the LAR, as supplemented. The licensee adequately applied the guidance for establishing PRA technical acceptability for the models. The NRC staff further considered the key assumptions and sources of uncertainty identified by the licensee, the proposed use of surrogates in the PRA models for evaluating RICTs for specific TS functions, and credit for FLEX. Therefore, the NRC staff finds that the scope and technical acceptability of the IEPRA and FPRA are commensurate with the RICT application for use in the integrated decision-making process and consistent with RG 1.174.

Evaluation of Seismic Hazard

The licensee's proposed approach for including the seismic risk contribution in the RICT calculation is to add a seismic CDF penalty and a seismic LERF penalty to each RICT calculation. The proposed seismic CDF penalty is based on using the plant-specific seismic hazard curves developed in response to the Near-Term Task Force Recommendation 2.1 as noted in the letter from Entergy dated March 27, 2014 (Reference 25), a plant-level high confidence of low probability of failure (HCLPF) capacity of 0.15g referenced to peak ground acceleration (PGA), and a composite beta factor of 0.4 to represent the uncertainty parameter for seismic capacity. Using these inputs, the licensee calculated a seismic CDF penalty of 4.24E-06 per year. The NRC staff finds the proposed method to determine the seismic CDF acceptable because it is consistent with the approach used in NRC Generic Issue (GI)-199 , "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated September 2, 2010 (Reference 26). The NRC staff performed an independent convolution using the input parameters identified by the licensee to confirm the proposed seismic CDF penalty.

The proposed seismic LERF penalty is based on a convolution of the estimated seismic CDF, as described above, with a limiting fragility for containment isolation failure, which was also assumed to be an HCLPF of 0.15g referenced to PGA. Using these inputs, the licensee calculated a seismic LERF penalty of 1.94E-06 per year. The NRC staff finds the proposed method to determine the seismic LERF penalty acceptable because the use of a 0.15g PGA HCLPF as the fragility for containment isolation is conservative.

The licensee did not address the incremental risk associated with seismic-induced LOOP in the LAR, as supplemented. The NRC staff used a typical LOOP fragility and site-specific seismic hazard curve to estimate the seismic-induced LOOP frequency at a level of approximately 1E-5 per year. This frequency is approximately 1 percent of the total internal events 24-hour non-recovered LOOP frequency at a level of 1E-3 per year already addressed in the internal events PRA model. Therefore, the NRC staff finds that the exclusion of seismic-induced LOOP frequency from the non-recovered LOOP frequency in the internal events PRA model has an insignificant impact on the RICT program calculations.

The NRC staff finds that, during RICTs for SSCs credited in the design-basis to mitigate seismic events, the licensee's proposed methodology captures the risk associated with seismically induced failures of redundant SSCs, because such SSCs are assumed to be fully correlated. In summary, the NRC staff finds the licensee's proposal to use the seismic CDF contribution of 4.24E-06 per year and a seismic LERF contribution of 1.94E-06 per year acceptable for the licensee's RICT program for Waterford 3 because (1) the licensee used the most current site-specific seismic hazard information for Waterford 3, (2) the licensee provided justification to use a plant-level HCLPF value of 0.15g, which is higher than the value of 0.1g used in the GI-199 evaluation, and kept the same composite beta factor of 0.4 used in the GI-199 evaluation, (3) the licensee determined a seismic LERF penalty based on its estimate of seismic CDF combined with a containment isolation fragility of 0.15g PGA HCLPF, and (4) the licensee will add the baseline seismic risk to RICT calculations with an assumption of fully correlated failures, which is conservative for SSCs credited in seismic events, while any potential for non-conservative results for SSCs that are not credited in seismic events is small or nonexistent.

Evaluation of Other External Hazards

The licensee evaluated external hazards in table E4-1, “External Hazard Evaluation,” in enclosure 4 to the LAR. This table was subsequently revised in the LAR supplement dated May 16, 2022. For the external flooding hazard, the licensee concluded that this hazard has an insignificant contribution to risk based on updated plant data, flood history, and new measures for risk management as well as the focused evaluation for external flooding using “Waterford Steam Electric Station, Unit 3 - Flood Focused Evaluation Assessment (CAC No. MF9710; EPID L-2017-JLD-0009),” (Reference 27). The NRC staff reviewed the licensee’s considerations of external flooding hazards for Waterford 3 and finds that the external flooding hazard has an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs, which is consistent with the conclusion of the NRC staff’s review of the licensee’s flood focused evaluation assessment.

For the extreme winds and tornados hazard, the licensee stated that this hazard has an insignificant contribution to risk based on the plant being designed for extreme winds and tornado loadings that are substantially higher than those required from the plant’s licensing basis and the thickness of tornado missile barriers protecting safety-related SSCs. In addition, the licensee stated that extreme winds and tornados are considered for the initiating events analysis for the PRA model for a LOOP. The NRC staff reviewed the licensee’s evaluation of the risk from extreme winds and tornados and finds that it is acceptable for the RICT program because the impacts from the extreme winds and tornado are considered in the PRA model as a LOOP event.

For the other external hazards evaluated in table E4-1 in enclosure 4 to the LAR, the licensee concluded that these external hazards have an insignificant contribution to risk and proposed that these external hazards be screened out from the RICT program. The other external hazards evaluated in table E4-1 are those identified for consideration in non-mandatory appendix 6-A of the ASME/ANS RA-Sa-2009 PRA standard , which provides a guide for identification of most of the possible external events for a plant site. The NRC staff reviewed the licensee’s evaluation of other external hazards and finds that the licensee’s consideration of risk from other external hazards is acceptable because the screening criteria used in table E4-1 are the same as the criteria presented in supporting requirements for screening external hazards EXT-B1 and EXT-C1 of the ASME/ANS RA-Sa-2009 PRA standard.

In summary, the NRC staff finds that the contributions from external flooding, extreme winds and tornados, and other external hazards have an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs because they either do not challenge the plant or they are bounded by the external hazards analyzed for the plant.

Application of PRA Models, Results, and Insights in the RICT Program

At the time of the submittal of the LAR and the LAR supplement dated April 8, 2021, the licensee had not fully completed the development of its CRMP tool (hereinafter referred to as the real-time risk (RTR) model). In response to NRC staff RAI 03, the licensee described the activities planned for completing the RTR model. The licensee described the planned verification and validation and benchmarking activities, which include comparison with prior models, and results from the maintenance rule models. The licensee also described its process for the RTR model update. The licensee tracks any changes to the PRA or the plant (including engineering changes, procedure revisions, licensing revisions, model improvement, plant-specific data changes, and industry research) and assesses their impact on the model. The licensee described its criteria for performing planned updates to the RTR model and interim updates to the RTR model when significant model changes occur that would impact the RICT calculations.

In its response to NRC staff RAI 07, the licensee described its process for deriving RMAs, which includes predetermined RMAs for procedures, RTR model insights, and operator experience, as well examining the list of important trains provided by the RTR model.

Based on the above, the NRC staff finds that the Waterford 3 PRA models and RTR model used will continue to reflect the as-built, as-operated plant consistent with RG 1.200, Revision 2 for ensuring PRA acceptability is maintained. Therefore, the NRC staff concludes that the proposed application of the Waterford 3 RICT program is appropriate for use in the adoption of TSTF 505 for performing RICT calculations.

In enclosure 5, “Baseline CDF and LERF,” to the LAR, as supplemented, the licensee provided the estimated total CDF and LERF of the base PRA models to demonstrate that Waterford 3 meets the 1E-4/year CDF and 1E-5/year LERF criteria of RG 1.174 consistent with the guidance in NEI 06-09-A and that these guidelines will be satisfied for implementation of a RICT program.

The licensee has incorporated NEI 06-09-A into TS 6.5.19. The estimated current total CDF and LERF for Waterford 3 PRAs meet the RG 1.174 guidelines; therefore, the NRC staff concludes that the PRA results and insights to be used by the licensee in the RICT program will continue to be consistent with NEI 06-09-A.

Tier 1: Conclusions

Based on the above, the NRC staff finds that the licensee has satisfied the intent of Tier 1 for determining the PRA acceptable, and that the scope of the PRA models (i.e., IEPRA and FPRA) and other evaluated hazards, external hazards, and seismic methodology is appropriate for this application.

Tier 2: Avoidance of Risk-Significant Plant Configurations

Tier 2 evaluates the capability of the licensee to identify and avoid risk-significant plant configurations that could result if equipment, in addition to that associated with the proposed change, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The limits established for entry into a RICT program and for RMA implementation are consistent with the guidance of Nuclear Utility Management and Resources Council (NUMARC) 93-01, Revision 4F, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” dated April 2018 (Reference 28) endorsed by RG 1.160, Revision 4, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” dated August 2018 (Reference 29), as applicable to plant maintenance activities. The LAR also explains that RMAs will be implemented, in accordance with current plant procedures, no later than the time at which the 1E-06 ICCDP or 1E-07 ICLERP threshold is reached and under emergent conditions when the instantaneous CDF and LERF thresholds are exceeded.

The NRC staff concludes that the RICT program requirements, that includes limits established for entry into a RICT, and implementation of RMAs are consistent with NEI 06-09-A. Therefore, the NRC staff finds that the proposed changes are consistent with the intent of Tier 2.

Tier 3: Risk-Informed Configuration Risk Management

Tier 3 stipulates that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. The proposed RICT program establishes a CRMP based on the underlying PRA models. The CRMP is then used to evaluate configuration-specific risk for planned activities associated with the RMTS extended CT, as well as emergent conditions that may arise during an extended CT. This required assessment of configuration risk, along with the implementation of compensatory measures and RMAs, is consistent with the principle of Tier 3 for assessing and managing the risk impact of out of service equipment.

Paragraph 50.36(c)(5) of 10 CFR identifies administrative controls as “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.” In enclosure 8, “Attributes of the CRMP Model,” to the LAR, the licensee confirmed that future changes made to the baseline PRA models and changes made to the online model (i.e., CRMP) are controlled and documented by plant procedures. In enclosure 10, “Program Implementation,” to the LAR, the licensee identified the attributes that the RICT program procedures will address, which are consistent with NEI 06-09-A. In response to APLA RAI 03.b in the LAR supplement dated May 16, 2022, the licensee committed to update licensee procedure EN-DC-151, “PRA Maintenance and Update,” to include criteria to perform an interim update to the PRA models used in the RTR model specific to the RICT application. The NRC staff finds that the licensee has identified appropriate administrative controls consistent with NEI 06-09-A and that it will continue to meet 10 CFR 50.36(c)(5).

Based on the licensee’s incorporation of NEI 06-09-A in the TSs, as discussed in LAR attachment 1; use of RMAs as discussed in LAR enclosure 12, “Risk Management Action Examples”; and because the proposed changes are consistent with the Tier 3 guidance, the NRC staff finds the licensee’s Tier 3 program is acceptable and supports the proposed implementation of the RICT program.

Key Principle 4: Conclusions

The licensee has demonstrated the technical acceptability and scope of its PRA models and alternative methods, including consideration of the impact of seismic events, non-seismic external hazards, and other hazards, and that the models can support implementation of the RICT program for determining extensions to CTs. The licensee has made proper consideration of key assumptions and sources of uncertainty. The risk metrics are consistent with the approved methodology of NEI 06-09-A and the acceptance guidance in RGs 1.174 and 1.177. The RICT program will be controlled administratively through plant procedures and training and follows the NRC-approved methodology in NEI 06-09-A. Based on the above, the NRC staff concludes that the RICT program satisfies the fourth key principle and is, therefore, acceptable.

Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring

RGs 1.174 and 1.17 establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. Enclosure 11, “Monitoring Program,” to the LAR states that the SSCs in scope of the RICT program are also in scope of 10 CFR 50.65 for the Maintenance Rule. The Maintenance Rule monitoring programs will provide for evaluation and disposition of unavailability impacts, which may be incurred from implementation of the RICT program. Furthermore, in enclosure 11 to the LAR, the licensee confirmed that the cumulative risk is calculated at least every refueling cycle, but the recalculation period does not exceed 24 months, which is consistent with NEI 06-09-A.

The NRC staff concludes that the RICT program satisfies the fifth key principle because: (1) the RICT program will monitor the average annual cumulative risk increase as described in NEI 06-09-A, thereby ensuring that the program, as implemented, continues to meet guidance in RGs 1.174 and 1.177 guidance for small risk increases; and (2) all affected SSCs are within the Maintenance Rule program, which is used to monitor changes to the reliability and availability of those SSCs.

3.2 Optional Changes and Variations from TSTF-505

The NRC staff evaluated the proposed use of RICTs in the optional changes and variations discussed above in section 2.0 in conjunction with evaluating the proposed use of RICTs in each of the individual LCOs, Required Actions, and CTs. The NRC staff's evaluation of the licensee's proposed use of RICTs in the variations against the key safety principles is discussed above. Based on the above section, the NRC staff finds that each of the five key principles have been met and concludes that the proposed optional changes and variations are acceptable

3.3 Conditions Requiring Additional Technical Justification

3.3.1 Additional Technical Justification Specified in TSTF-505

Table 1 of TSTF-505 contains a list of LCO conditions that may be proposed for inclusion in the RICT program where additional information may be necessary to explain why the condition would not represent a loss of specified safety function as used in the RICT program. Suggestions are provided in the table, but the suggestions may not be all encompassing for all plants. Licensees should provide sufficient information when adopting the listed Required Actions to justify that the condition does not represent a loss of specified safety function as used in the RICT program.

The following information was obtained from the LAR and UFSAR, related to the containment and plant systems TSs where TSTF-505 specified the need for additional technical justification:

STS 3.6.6A "LCO Containment Spray and Cooling Systems Atmospheric and Dual)"

As indicated in table E1-1 of enclosure 1 to the LAR, the containment spray and cooling systems are treated separately under TS LCOs 3.6.2.1, "Containment Spray System," and 3.6.2.2, "Containment Cooling System." The conditions proposed for risk-informed allowed outage times involve inoperability of only one of two trains for each system, and the two conditions may be entered simultaneously. This consideration is consistent with the design-basis described in Section 6.2.2, "Containment Heat Removal Systems," of the Waterford Unit 3 UFSAR, which assumes the operation of one spray and one cooling train providing containment cooling for accident mitigation.

Section 2.3, "Optional Changes and Variations," of attachment 1 to the LAR provides the following information addressing TS LCO 3.6.2.1 and TS LCO 3.6.2.2:

> NUREG 1432 includes a single technical specification for Containment Spray and Cooling Systems. Waterford 3 has individual TS for each system. The

> cooling function of the systems is redundant. However, iodine removal is a function of the Containment Spray system but not the Containment Cooling System. The Waterford 3 RICT program will include a completion time for a single Containment Spray train inoperable. The program will also include a completion time for one of the two trains of Containment Cooling being inoperable. Having both Containment Spray trains inoperable is a loss of iodine removal function and will not be included in the RICT program.

Thus, the operability of one spray train ensures that the iodine removal function would be maintained. These systems are explicitly modeled in the PRA, and the PRA success criteria are consistent with the design-basis. Therefore, the proposed actions would not present a loss of function condition for the containment cooling or iodine removal functions.

STS 3.7.2.A, LCO [Two] MSIVs shall be OPERABLE "One MSIV inoperable in MODE 1"

As indicated in table E1-1 of enclosure 1 to the LAR, the MSIVs are explicitly modeled in the PRA. The PRA success criteria are consistent with the design-basis. The Waterford 3 MSIVs are double-disk gate valves that prevent flow in either direction. Thus, closure of one valve, as required for DBAs involving a loss of main steam line pressure boundary integrity, would result in isolation of one SG from the break, consistent with the design-basis. Therefore, the proposed action would not present a loss of function condition for the SG isolation function.

3.3.2 Plant-Specific LCO Variations from TSTF-505

Two plant-specific LCOs and associated Actions for which the licensee is proposing to apply the RICT program that are variations from the STSs in NUREG-1432 and the evaluated actions in TSTF-505 were identified in enclosure 13, "Waterford 3 to Standard Technical Specification Cross Reference," to the LAR. The NRC staff found the following TS LCO Actions to differ significantly from those considered for inclusion in TSTF-505:

- LCO 3.7.1.2, Action 'd' for an inoperable EFW system that remains capable of delivering at least 100 percent flow to either SG.

- LCO 3.7.1.7, Action 'a' for an inoperable ADV automatic actuation channel.

The NRC staff evaluated the information provided for these plant-specific LCOs and associated Actions to confirm that the condition does not represent a TS loss of function. Acceptability is based on appropriate constraints to preclude application of a RICT when a loss of function condition exists or if the system functions are not appropriately modeled in the PRA to reflect the risk associated with a loss of redundancy.

For conditions under LCO 3.7.1.2, Action 'd,' which applies for an inoperable EFW system that remains capable of delivering at least 100 percent flow to either SG, the safety function is maintained because, as specified in the Action, adequate flow can be delivered to either SG for decay heat removal and accident mitigation functions. TSTF-505 includes consideration of a RICT for STS LCO 3.7.5.B, which applies when one auxiliary feedwater train is inoperable for reasons other than a turbine-driven pump inoperable as a result of one inoperable steam supply. This condition is similar to the proposed conditions under TS 3.7.1.2, Action 'd,' which applies when the EFW system can deliver 100 percent flow to each SG. When 100 percent flow can be delivered to each SG with operable EFW equipment, equipment equivalent to at least one train of EFW is operable. In addition, the bases for TS LCO 3.7.1.2, Action 'd,' describes that the action applies to conditions including an inoperable turbine-driven pump, both motor-driven pumps inoperable, or one of two redundant flow paths. Therefore, these conditions are equivalent to the STS LCO 3.7.5.B accepted for a RICT in TSTF-505. In the LAR supplement dated May 16, 2022, the licensee provided estimated RICTs for an inoperable turbine-driven pump and two inoperable motor-driven pumps in table E1-2, which supports evaluation of the PRA model of the system. Therefore, there is no loss of function condition associated with the condition, and a RICT is appropriate.

LCO 3.7.1.7 Action ‘a’ applies when one ADV automatic actuation channel is inoperable. The accident analyses credit automatic ADV actuation at power levels above 70 percent of full power to mitigate certain DBA conditions, and the success criterion is that one of two ADVs actuate to remove heat from the reactor coolant system through the SGs. The action ensures that one ADV actuation channel is operable and, therefore, represents only a loss of redundancy. In the LAR supplement dated May 16, 2022, table E1-1 indicates that the PRA model includes automatic actuation of the valves and the PRA success criteria are consistent with the design-basis. Therefore, there is no loss of function condition associated with the condition, and a RICT is appropriate.

3.4 Changes to the Operating License

In the LAR and in the licensee’s responses to NRC staff RAIs there were certain specific actions that the NRC staff identified as being necessary to support the conclusion that the implementation of the proposed RICT program met the requirements of the RICT. The NRC staff finding on the acceptability of the implementation of the RICT program for the TS LCOs in this SE is dependent on the completion of the following license condition.

In the LAR supplement dated August 19, 2022, the licensee proposed the following changes to the Waterford 3 renewed facility operating license:

> Entergy will complete closure of the four Human Reliability Analysis (HRA) Finding level Facts and Observations (F&Os) identified as Finding Numbers HR 1-2, HR 7-1, HR 7-3, and HR 7-4 in Table A3-2 of Entergy letter to NRC, dated April 25, 2022, and in Table E2-2 of Entergy letter to NRC, dated May 16, 2022, using an accepted NRC process (Nuclear Energy Institute (NEI) Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13) prior to implementation of 10 CFR 50.69 and the risk-informed completion time (RICT) program.

The NRC staff notes that prior approval would be required for a change to the RICT program, or the implementation of the RICT program as described in TS 6.5.19, and the implementation item in the LAR supplement dated May 16, 2022.

The NRC staff finds that the above license condition is acceptable because it adequately implements the RICT program using models, methods, and approaches consistent with applicable guidance that are acceptable to the NRC. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the license condition, with the expectation that any issues discovered during this review, or concerns with regard to its adequate completion, would be tracked and dispositioned appropriately under the licensee’s corrective action program and could be subject to appropriate NRC enforcement action.

3.5 Technical Evaluation Conclusion

The NRC staff has evaluated the proposed changes against each of the five key principles in RGs 1.174 and RG 1.177, including the optional variations from the approved TSTF 505. The NRC staff concludes that the proposed changes satisfy the key principles of risk-informed decision-making identified in RG 1.174 and RG 1.177 and, therefore, the requested adoption of the proposed changes to the TSs, implementation items, and associated guidance, is acceptable to the NRC staff to ensure that the Commission's regulations continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment on January 25, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 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 which was published in the Federal Register on May 18, 2021 (86 FR 26954), that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

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

7.0 REFERENCES

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Principal Contributors: J. Evans
M. Biro
D. Wu
S. Alferink
E. Kleeh
V. Goel
M. Li
J. Wilson
S. Jones

Date: February 17, 2023

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT NO. 270 RE: ADOPTION OF TSTF-505, REVISION 2, “PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4b” (EPID L-2021-LLA-0014) DATED FEBRUARY 17, 2023

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JWilson, NRR
KWest, NRR

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